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Generation IV Roadmap

R&D Scope Report for Liquid-Metal-Cooled Reactor Systems

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MEMBERS OF THE LIQUID-METAL-COOLED REACTOR SYSTEMS TECHNICAL WORKING GROUP

Stephen Rosen	Co-Chair	South Texas Project (retired), United States
Yutaka Sagayama	Co-Chair	Japan Nuclear Cycle Development Institute, Japan
Michael Lineberry	Technical Director	Angonne National Laboratory, United States
Gerhard Bart		Paul Scherrer Institut, Switzerland
Charles Boardman		General Electric (retired); consultant, United States
Jean-Louis Carbonnier		Commissariat à l'Energie Atomique, France
Luciano Cinotti		EC-Ansaldo, Italy
Jean-Paul Glatz		EC, Centre Commum de Recherch, Germany
Orlando J. A. Goncalves-Filho		Instituto de Engenharia Nuclear/CNEN, Brazil
Do Hee Hahn		Korea Atomic Energy Research Institute, Korea
Masakazu Ichimiya		Japan Nuclear Cycle Development Institute, Japan
John Lee		University of Michigan, United States
Ning Li		Los Alamos National Laboratory, United States
Claes Nordborg		Organisation for Economic Co-operation and Development
Ronald Omberg		Pacific Northwest National Laboratory, United States
Alexander Stanculescu		International Atomic Energy Agency, UN
Kune Y. Suh		Seoul National University, Korea
John Tuohy		Burns & Roe, United States
David Wade		Argonne National Laboratory, United States

OTHER CONTRIBUTORS

Todd Allen	RIT Representative	Argonne National Laboratory, United States
Ralph Bennett	RIT Representative	Idaho National Engineering and Environmental Laboratory, United States
Ted Marston	GRNS Representative	Electric Power Research Institute, United States
Luis Nunez	DOE Representative	Argonne National Laboratory, United States

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EXECUTIVE SUMMARY

In this report the Generation IV Technical Working Group 3 (TWG 3) on Liquid Metal Reactors presents its R&D plan for development of selected Gen IV concepts under its purview. The concepts (in fact, sets of concepts) considered here are those that emerged as the most promising during the many months of screening and evaluation.[a]

The activity reported on here by TWG 3 goes well beyond mere selection of a small set of reactor systems and fuel cycles, and subsequent specification of the R&D needed to bring them to fruition. If successfully executed the TWG 3 R&D plan would open up the possibility for nuclear power to make a huge contribution to global electricity supply for centuries. This of course refers to the capability of TWG 3 reactors to create more fissile material than they consume, a characteristic brought about because of the high-energy neutron spectrum of all TWG 3 systems, so-called “fast reactors”.

These reactors, however, will be deployed in a world of thermal reactors, and only when this realization is taken into account can the full potential of the entire future nuclear energy system be reached. As the number of deployed fast reactors grows in the years after initial deployment, i.e., decades from now, it is conceivable that they will be operating in an environment of multiple types of thermal reactors. Regardless of the details, the fast reactors will operate in *symbiosis* with the thermal reactors. But the specific symbiotic arrangement, and how far the arrangement goes in meeting all of the Gen IV aspirations, is very dependent on the R&D plans that are laid today.

Consider fuel recycle, which must be done in order for fast reactors to be practical. TWG 3 undertook the R&D planning for fuel processing, fabrication, and waste management technologies, with greater emphasis than that given to the reactor systems. Just as the Gen IV goals for reactor technologies are ambitious, so too are the goals and expectations for the fuel recycle technologies. As an example, all fuel cycle technologies considered by TWG 3 included a goal of recycling >99.9% of the actinides, mainly because this eases the technical requirements on future waste repositories. A number of the sustainability criteria for Gen IV systems recognized the significance of the waste management benefits of actinide recycling (and destruction in the reactors). Clearly though, there is little waste management benefit if the high-level wastes from the fast reactors are free of actinides, while the many thermal reactors are sending their actinides to the repositories. To realize the full potential that the system offers, the symbiosis centers on the fuel cycle potential. Processing some amount of thermal reactor spent fuel will be necessary to provide the initial fuel inventory of the fast reactors. But beyond that, essentially all of the thermal reactor spent fuel must be eventually recycled, with technologies complementary to those used in the fast reactor processing, in order to achieve the waste-related sustainability goals adopted for Gen IV.

The advanced fuel cycle technology for the fast reactors has therefore been planned and developed with consideration for the fuels and fuel cycle needs of the thermal reactors that will be deployed alongside them. Fuel cycle and fuels technologies thus occupy the center stage of the TWG 3 R&D plan. This is in concert with the recommendations of the Gen IV Roadmap NERAC Subcommittee (GRNS). However, TWG 3 also recommends that for the sodium-cooled reactors, while perhaps de-emphasized in the near term, some development work should continue on the reactor technologies. This is particularly

a. This scope report was completed over a several-month period toward the end of the Gen IV R&D roadmap activity, through a number of iterations. Concurrent with this iterative process, a number of decisions were made by Gen IV leadership groups that resulted in the six specific concept sets that were ultimately chosen for further study. TWG 3 made an early decision about which concept sets to include in the Scope Report. Fortunately, the concept sets chosen by TWG 3 turns out to be very close to those ratified by the Gen IV International Forum in July 2002.

important in certain aspects of reactor design aimed at cost reduction, reactor safety, and nuclear security. For the lead-based systems, TWG 3 recommends a "science-based" R&D program that aims to resolve fundamental issues first, before emphasizing specific concept selection.

In this report we thus develop integrated R&D plans for two broad classes of reactor systems and their associated fuel cycles. They were derived from four concept sets that emerged from the screening-for-potential and final evaluation rounds. Each integrated R&D plan has multiple pathways or branches. Consider the sodium-cooled set of concepts, where by far the majority of the specific concepts fall into two applications:

1. Medium-to-large oxide fueled reactors with advanced aqueous processing and remote fabrication (the L1 concept set, using the shorthand of the Roadmap Integration Team)
2. Medium sized metal fueled reactors with pyroprocessing and remote fabrication (denoted L2).

Note that early in the roadmap, TWG 3 members from Japan wanted a metal fuel option preserved for their large JNC Sodium-Cooled Fast Reactor (JSFR) system. By producing an integrated R&D plan with distinct paths, such choices are facilitated. Also, there are numerous areas (cost reduction strategies, in-service inspection and repair technologies, nuclear security) where the R&D activities are nearly identical for the two concept sets. Finally, an integrated plan may help foster international collaboration and exchange.

For somewhat different reasons, TWG 3 decided to construct an integrated R&D plan for the selected concept sets cooled with lead and lead-bismuth eutectic (LBE), which comprise concept sets L6 and L4. However, the specific applications in these sets take on a much broader range than L1 and L2, and this is especially true for L6. However, the specifics of the application matter much less in L6/L4, because the state of the technology for these systems is at a considerably earlier stage than that for sodium-cooled systems. There are a number of basic development questions common to all the lead-based systems. These need to be addressed first, with emphasis on specifics of concept design reduced in the meantime. This gives rise to what TWG 3 has termed a "science based" program for the lead-based systems, and it is what the group recommends and has attempted to develop in this plan.

The emphasis is on "viability R&D" throughout, i.e., those developments that are necessary to resolve before a significant engineering application would be expected to be undertaken. However, the end point for the Generation IV Roadmap remains what it was at the beginning, the completion of a credible conceptual design for both reactor and fuel cycle facilities. We generally show those design activities in the plan, though in considerably less detail than for the items considered as "viability R&D".

In the development of the R&D plan, the TWG 3 took care to describe the rationale for the concept and the function it is best suited to perform in the Gen IV suite of concepts, as well as the specific technology gaps that must be resolved for the concept to meet its potential. Knowing the concept rationale helps to place it correctly in the Gen IV Roadmap to attain a global energy supply meeting Gen IV goals.

In the simple table below we list the main distinguishing features of the concept sets.

Distinguishing Features of the TWG 3 Concept Sets

Concept Set	Features
L1	Large-sized, sodium-cooled reactors, oxide-fueled, advanced aqueous/dry process and advanced refabrication technologies
L2	Medium-sized, sodium-cooled reactors, metal alloy fueled, pyroprocess fuel cycle technology
L4	Lead Bismuth Eutectic (LBE) cooled, medium-size reactors, metal or nitride fueled, pyroprocess fuel cycle technology
L6	Small reactors, lead or LBE cooled, generally with cartridge refueling, generally pyroprocess fuel cycle technology. Sometimes referred to as "battery concepts"

It is the TWG 3 view that nations with major nuclear power programs need not choose between sodium and lead coolant, and might choose to develop both. A sodium option might be selected because of the substantial technology base as well as new potential that it holds. A lead-based option might be pursued for the promise it holds in safety and economics, and in the potential to operate in much higher temperature regimes.

These two groups of concepts serve different missions, have different market niches, and are at substantially different states of readiness and therefore have considerably different deployment times.

Concepts in L1/L2 are suited especially for industrialized nations having substantial technical and institutional supporting infrastructure and large electrical grids. The concepts in L1/L2 evolved from the traditional lines of fast reactor development, the result of more than four decades of national programs in France, the U.K., Japan, the U.S., Germany, and Russia. The missions that this concept set is targeted for include large-scale electricity production, management of wastes, from other reactors as well as their own, and creation of fissile material (for other reactors as well as their own).

L6/L4 (especially L6) has been specifically tailored to the needs of developing countries, having small grids, small incremental deployments as constrained by capital financing availability, and/or limited technical and institutional infrastructure. Further, such countries may have little interest in or capacity for development of indigenous front and back end fuel cycle infrastructure.

Also, there are substantial differences in the range of applications the reactors would serve in a symbiotic nuclear energy system comprised of Gen IV fast and thermal spectrum reactors. L1/L2 power plants can function either as net transuranic burners or as breeders *with short doubling times*, whereas L6/L4 are better suited as burners or fissile self-sufficient plants, with a capability for no more than *long doubling time*. All the TWG 3 concept sets will likely be configured as burners or fissile-self-sufficient systems in the early decades of Gen IV deployment, but only L1/L2 can transition eventually to short doubling time "fuel factories," producing excess fissile material needed to fuel other reactors in the overall system.

The innovations in concept set L6/L4 are broader in scope than those contained in L1/L2. Innovations include attempts to provide new functions from nuclear energy (expanding the products to include process heat and hydrogen production, perhaps, among others). Entry into new markets is possible (battery-sized plants); as is use of new fuel cycles (thorium cycles, nitride fuels). It appears possible to couple to new energy converters (supercritical steam Rankine, supercritical CO_2 Brayton; driving chemical plants and/or chemical heat pumps), and to employ new heat transport schemes. And from a fuel cycle perspective, it is possible to develop new recycle flowsheets (nitride-based cycles, thoruim-based cycles, dispersion fuels).

The main technology gaps are, first for the sodium-cooled systems:

- Fuel cycle
  - Advanced aqueous process: demonstration of high actinide recoveries, proliferation resistance features, ability to meet economic goals, and remote fabrication technology
  - Pyroprocess: demonstration of plutonium and minor actinide extraction at larger scale, high actinide recoveries, minimization of secondary streams, certification of high-level waste forms.
- Fuel development
  - For all fuel types, irradiation and transient testing of recycled fuel fabricated with prototypic equipment
  - For metal fuels, limited transient testing at high burnup.
- Reactor R&D
  - Capital cost reduction based on, for example, design innovations or modularization
  - In-service inspection and repair.
- Reactor safety
  - Demonstration of passive safety design
  - Accommodation of extremely low probability but higher consequence accident scenarios.

For the lead-cooled systems:

- The main science-based program issues
  - Coolant compatibility
  - Achieving high operating temperatures.
- Fuel development
  - Steady-state irradiation performance and transient testing.
- Fuel Cycle
  - Nitride fuel recycle, with recovery and recycle of N-15
  - Basic flowsheet development, then experiments up to appropriate scale.
- Reactor Safety
  - Passive safety assurance in the design basis
  - Treatment of beyond-design-basis scenarios.

The above technology gaps form the basis for the R&D plans that are developed in significant detail in Section 3 (sodium-cooled systems) and Section 4 (lead-cooled systems). R&D Summary sheets are provided in Appendix A of this report.

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ACRONYMS

AA	advanced aqueous
AAA	advanced accelerator applications
ABLE	axial blanket partially eliminated
ADS	accelerator driven systems
ANL	Argonne National Laboratory
ATWS	anticipated-transient-without-scram
BDBAs	beyond design basis accidents
CDAs	core disruptive accidents
CFD	computational fluid-dynamics
DBAs	design basis accidents
DFR	Dounreay Fast Reactor
EBR-II	Experimental Breeder Reactor-II
EMG	Evaluation Methodology Group
FAIDUS	fuel assembly with an inner duct structure
FCCI	fuel-cladding chemical interaction
FCF	Fuel Conditioning Facility
FCCG	Fuel Cycle Crosscut Group (Gen IV)
FFTF	Fast Flux Test Facility
GEM	gas expansion module
GRNS	Gen IV Roadmap NERAC Subcommittee
HE-N_2	highly enriched (in N-15) nitrogen
HFEF	Hot Fuel Examination Facility
HMLC	heavy metal liquid coolant
I/P	initiating phase
INEEL	Idaho National Engineering and Environmental Laboratory
IPPE	Institute of Physics and Power Engineering
IFR	Integral Fast Reactor
JAERI	Japan Atomic Energy Research Institute
JNC	Japan Nuclear Cycle Development Institute
JSFR	JNC Sodium-cooled Fast Reactor
KAERI	Korean Atomic Energy Research Institute
LBE	lead-bismuth eutectic
LME	liquid metal embrittlement

LTA	lead test assembly
LWR	light water reactor
MA	minor actinide
NERAC	(DOE) Nuclear Energy Advisory Committee
NPP	nuclear power plant
ORNL	Oak Ridge National Laboratory
PFR	Prototype Fast Reactor
PIEs	postulated initiating events
PRW	pressurized resistance welding
PSI	Paul Scherer Institute
RETF	Recycle Equipment Test Facility
RIAR	Research Institute of Atomic Reactors
RIT	Roadmap Integration Team
SG	steam generator
SASS	self-actuated shutdown system
T/P	transition phase
TREAT	Transient Reactor Test Facility
TRP	Tokai Reprocessing Plant
ULOF	unprotected loss of flow
ULOHS	unprotected loss of heat sink
UNH	uranyl nitrite hexahydrate
UT	ultrasonic testing
UTOP	unprotected transient over power

R&D Scope Report for Liquid-Metal-Cooled Reactor System

1. INTRODUCTION AND CONCEPT SET DESCRIPTION

In this report, Technical Working Group 3 (TWG 3) on Liquid Metal Reactors presents its R&D plan for development of the Generation IV concepts that emerged as the most promising in the many months of screening and evaluation. As is the case throughout Generation IV, it is concept sets rather than single concepts that remain as the roadmap work approaches its completion.

In this the last round of Generation IV Roadmap activity, i.e., the construction of the R&D plans, the working group has chosen to group the remaining concepts in its purview differently than in previous rounds. First, what was termed concept L3 (the 4S reactor concept), and concept set L5 (the large lead-cooled systems), are not being carried into this round of R&D planning (these "L" designations will be explained more below).[b]

Further, for several reasons TWG 3 chose to combine the remaining <u>four</u> concepts sets that emerged from the screening-for-potential and final evaluation rounds into <u>two</u> "integrated" R&D plans, each with multiple paths or branches. Consider, for example, the sodium-cooled set of concepts, where by far the majority of the specific concepts fall into two applications:

1. Medium-to-large oxide fueled reactors with advanced aqueous processing and remote fabrication (the L1 concept set)
2. Medium sized metal fueled reactors with pyroprocessing and remote fabrication (denoted L2).

Note, however, that early in the roadmap, TWG 3 members from Japan wanted a metal fuel option preserved for their large JSFR system. By producing an integrated R&D plan with distinct paths, such choices are facilitated. Also, there are numerous areas where the R&D activities are nearly identical for the two concept sets. Finally, an integrated plan may help foster international collaboration and exchange.

For somewhat different reasons, TWG 3 decided to construct an integrated R&D plan for the selected concept sets cooled with lead and lead-bismuth eutectic (LBE), which are concept sets L6 and L4. However, the specific applications in these sets take on a much broader range than L1 and L2, and this is especially true for L6. The specifics of the application matter much less in L6/L4, because the state of the technology for these systems is at a considerably earlier stage than that for sodium-cooled systems. There are a number of basic development questions common to all the lead-based systems. These need to be addressed first, with emphasis on concept design reduced in the meantime. This gives rise to what TWG 3 has termed a "science based" or "technology based" program for the lead-based systems, and it is what the group recommends and has attempted to develop in this plan.

The report is organized as follows. Descriptive material on L1/L2 and L6/L4 will be presented as part of this introductory section, but (once more) in this section only, as separate concept sets. Section 2 will provide a synopsis of the main technology gaps that the R&D programs must address. Section 3 then presents the detailed R&D plan for L1/L2. The narrative is generally structured so as to present in each subsection a brief and more detailed statement of the technology gap that the R&D in the subsection addresses. Section 4 presents the science-based R&D plan for the lead and LBE cooled systems.

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b. L5 was ultimately included in the final GIF selection.

Appendix A contains the summary R&D plan spread sheets, in the format suggested by the Roadmap Integration Team (RIT).

Considerable emphasis is placed on the fuel cycle requirements, particularly for L1/L2, because all TWG 3 systems rely on a closed fuel cycle. In addition, beyond the handful of plants that might use global stocks of plutonium already separated, the startup fuel for any TWG 3 systems must come from processing spent thermal reactor fuel. Moreover, capturing the overall Gen IV waste management benefits of the advanced processing options showcased in this report will require that thermal reactor spent fuels be processed with technologies that are complementary, if not identical, to those that TWG 3 recommends in this report. We thus address fuel cycle technology options, and R&D requirements, for coupling thermal reactor options to fast reactor concepts.

Lastly the emphasis is on "viability R&D" throughout, i.e., those developments that are necessary to resolve before a significant engineering application would be expected to be undertaken. However, the end point for the Generation IV Roadmap remains what it was at the beginning, the completion of a credible conceptual design for both reactor and fuel cycle facilities. We generally show those design activities in the plan, though in considerably less detail than for the items considered as "viability R&D".

In the simple Table 1 below we list the main distinguishing features of the concept sets. We have adopted the "L" shorthand notation used by the Gen IV Roadmap Integration Team (RIT), and therefore we also include descriptors used in earlier TWG 3 reports:

Table 1. Distinguishing features of the TWG 3 concept sets.

Concept Set	Features	Prior Designations
L1	Large-size, sodium-cooled reactors, oxide-fueled, advanced aqueous/dry process and advanced refabrication technologies	Group A or Track A
L2	Medium-size, sodium-cooled reactors, metal alloy fueled, pyroprocess fuel cycle technology	Group B or Track B
L4	Lead Bismuth Eutectic (LBE) cooled, medium-size reactors, metal or nitride fueled, pyroprocess fuel cycle technology	Group C or Track C/D or Domestic C
L6	Small reactors, lead or LBE cooled, generally with cartridge refueling, generally pyroprocess fuel cycle technology. Sometimes referred to as "battery concepts"	Group D or Track D or Battery D

An important rationale for fast reactors is the sustainability that they uniquely offer, from the perspective of resource utilization and waste management. Any fast reactor with "breeding ratio" (defined as the units of fissile material created per unit of fissile material consumed) slightly greater than unity (to overcome the small process losses) can utilize essentially all of the world's uranium resource, not the ~1% of it that Light Water Reactors (LWRs) can achieve. All of the systems in TWG 3 can be operated to produce electricity; all can be operated as most efficient net consumers of fissile material; and all can be operated to create net fissile material, though some TWG 3 concept sets are considerably better than others in this regard. In waste management, all the TWG 3 systems involve closed fuel cycles. Although they are considerably different in technical features, they all seek to recycle ~99.9% of the plutonium and minor actinides, and this considerably eases the technical demands placed on repositories, and can dramatically reduce the number of repositories that would be necessary. The Fuel Cycle Crosscut Group has recommended R&D on "integrated waste management" to exploit the opportunities for repository benefits resulting from recycle.

Some additional thoughts on the missions that these concept sets are aimed to serve might be worthwhile. It is the TWG 3 view that nations with major nuclear power programs need not choose between sodium and lead coolant, and might choose to develop both. A sodium option might be selected because of the substantial technology base as well as new promise that it holds. A lead-based option might be pursued for the potential it holds in safety and economics, and in the potential to operate in much higher temperature regimes.

Given that the technology status for L1/ L2 is advanced compared to L6/L4, and further that L1/L2 has a high expectation of meeting the Gen IV goals in sustainability and safety/reliability, it seems likely that a collection of strong functional discriminators and/or potential payoffs must be evident to justify Gen IV R&D investments in L6/L4 development. TWG 3 believes that such justification exists—as discussed below and in subsequent sections.

The first functional discriminator lies in targeted missions or market niches. L1/L2 are suited especially for industrialized nations having substantial technical and institutional supporting infrastructure and large electrical grids. L6/L4 (especially L6) has been specifically tailored to the needs of developing countries, having small grids, small incremental deployments as constrained by capital financing availability, and limited technical and institutional infrastructure. Further, such countries may have little interest in or capacity for development of indigenous front and back end fuel cycle infrastructure.

The second functional discriminator lies in the range of applications the reactors would serve in a symbiotic nuclear energy system comprised of Gen IV fast and thermal spectrum reactors. L1/L2 power plants can function either as net transuranic burners or as breeders *with short doubling times*, whereas L6/L4 are better suited as burners or fissile self-sufficient plants, with a capability for no more than *long doubling time*. Short doubling time requires high breeding ratio, short fuel cycle turnaround time out of the reactor, and high specific power (kw/kg fissile material). For economics reasons, high specific power translates to high power density (kw/liter) which translates to tight fuel-pin lattices and high coolant velocity. Early in the nuclear age it became clear that high specific gravity liquid metals such as lead or LBE could not support high power density and short doubling time because of pumping considerations—whereas sodium excelled in such service.

All the TWG 3 concept sets will likely be configured as burners or fissile-self-sufficient systems in the early decades of Gen IV deployment, but only L1/L2 can transition eventually to short doubling time "fuel factories", producing excess fissile material needed to fuel other reactors in the overall system. This would presumably occur as uranium ore grows scarce. L6/L4 plants would continue to function as fissile self-sufficient plants. Moreover, in the case of L6 plants, this is a crucial feature for achieving passive safety/passive load following in a long refueling interval deployment strategy.

In addition to the functional discriminators of differing targeted roles in the future nuclear energy system, L6/L4 offers numerous potential payoffs vis-à-vis L1/L2 through exploitation of the innate thermal/chemical/physical properties of lead and LBE as compared to sodium. These features include chemical inertness (with elimination of components in the heat transport circuits), high boiling point, low neutron slowing down power, and very low coolant absorption cross section.

The innovations introduced in concept L6/L4 are broader in scope than those in L1/L2. They include attempts to provide new functions from nuclear energy (expanding the products to include process heat and hydrogen production, perhaps among others); to enter new markets (battery-sized plants); to use new fuel cycles (thorium cycles, nitride fuels); to couple to new energy converters (supercritical-steam

Rankine cycles, supercritical CO_2 Brayton cycles; driving chemical plants and/or chemical heat pumps); to employ new heat transport schemes; and to develop new recycle flowsheets (nitride-based cycles, thorium-based cycles, dispersion fuels).

Taken together the L1/L2 and L6/L4 concept sets span a very broad range of application missions and through their coupling, with recycling, to the thermal-spectrum plants will provide actinide management for the entire Gen IV global energy supply system.

We next provide, for each of the four concept sets: (1) a brief concept description, including the baseline technology that was evaluated under this concept set, (2) a summary of the concept set's strengths and weaknesses in sustainability, (3) strengths and weaknesses in safety and reliability, and (4) strengths and weaknesses in economics.

1.1 Concept Set L1: Large Sodium-Cooled Reactors, Oxide Fuel, Advanced Aqueous Processing

Sodium cooled fast reactors in concept set L1 are medium to large in size (500 to 1500 Mwe), with mixed uranium-plutonium oxide fuel, supported by a fuel cycle based upon advanced aqueous reprocessing. Dry processing methods, distinct from the pyroprocess featured concept set L2, have been investigated in Russia and Japan, and will be included in the R&D planning as a possible pathway in L2/L1.

These concepts evolved from the traditional line of fast reactor development, the result of more than four decades of national programs in France, the United Kingdom, Japan, and the U.S., Germany, and Russia. The economic characteristics of these concepts are best suited to large, regulated, electrical markets where economies of scale offer a distinct competitive advantage. Industrial organizations capable of designing and constructing, as well as government agencies capable of regulating, these reactors exist in the major reactor development nations, both those cited as fast reactor development nations and others. The missions that this concept set is targeted for include large scale electricity production, management of wastes, and creation of fissile material. The specific concepts included in this concept set come from four of the traditional fast-reactor development countries (see Table 2).

Table 2. Concepts in Set L1: Large sodium-cooled, oxide fueled reactors; advanced aqueous process.

Concept	Known As	Size (Mwe)	Fuel	Outlet Temperature (°C)	Fuel Cycle	Country	Sponsoring Organization
M4	JSFR	1500	MOX	550	Advanced Aqueous (AA)	Japan	JNC
M6	BN-800	800	MOX	~550	AA/dry	Russia	IPPE Obninsk
M22	RNR-1500	1500	MOX	545	AA	France	CEA
M30	Compact Pool Fast Reactor	1500	MOX	545	AA?	U.K.	NNC Ltd.

The concepts in L1 are all based upon the desire to improve economics, safety, and component performance of this traditional line of sodium-cooled reactor development. An additional motivation is to ensure that advantages can be realized from future R&D, and therefore that this line of research does not become characterized as an archaic line of technology development. Actinide burning (i.e., management of wastes) is an example of a technology breakout that would provide one of the core missions for the concepts in L1, but which was not in their original design basis. Recent innovations in energy converters (e.g., supercritical CO_2 Brayton Cycles) often potential for significant cost reduction unforeseen in the original line of development. Narrowing the technology gaps will provide a distinct mission-ready leverage in that it would allow these concepts, which are relatively far along in market readiness, to fairly quickly penetrate the market in meaningful numbers.

The basis for the TWG 3 evaluation of L1 is primarily the JNC-sponsored concept from Japan, the JNC Sodium-Cooled Fast Reactor, or JSFR; and the concept from France, the CEA-sponsored RNR-1500. These are somewhat more recent than the Russian BN-800 concept, and there is considerably more design detail available for them than for either BN-800 or the entry from the U.K., the Compact Pool Fast Reactor.

In sustainability, the L1 concept set fully enjoys advantages of maximum resource utilization, minimum waste impacts, and reduced environmental impacts. Moreover, in some programs there is the intent or hope to go beyond the generic advantages of the modern new fuel cycle technology that is the baseline. For example, in the Japanese advanced aqueous process development, a goal is to go beyond U, Pu, and Minor Actinide (MA) co-extraction and further to separate long-lived fission products such as technetium, for subsequent transmutation in the reactor.

With regard to safety and reliability, in recent years great effort has gone into developing innovative design features that passively mitigate both Design Basis Accidents (DBAs) and Beyond Design Basis Accidents (BDBAs). Japan is placing significant emphasis on its “recriticality free” proposal, which seeks to assure through design features that no compaction-driven recriticality can occur even with low-probability events beyond the design base.

With respect to economics, the thrust in recent years has been to simplify the reactor and the fuel cycle design in order to achieve cost saving. In Japan, for example, this simplification takes the form of two-loop design for a 1500-Mwe plant, development of integral pumps and heat exchangers, and use of chrome-moly steels with high strength and low thermal expansion.

Table 3 shows the strengths and weaknesses of concept set L1.

1.2 Concept Set L2: Intermediate-Sized Sodium-Cooled Reactors, Metal-alloy Fueled, Pyroprocess Technology

Sodium cooled fast reactors in concept set L2 are intermediate in size (150 to 500 Mwe), with a uranium-plutonium-MA-Zr metal alloy fuel, supported by a fuel cycle based on the pyrometallurgical or “pyro” process. Like concept set L1, these concepts evolved from the traditional line of fast reactor development, especially that embodied by EBR-II in the U.S., but represent a point of departure from the traditional programs beginning in the early 1980s and continuing thereafter. The economic characteristics of these concepts are best suited to either large or intermediate-sized electricity markets, but through modularity or other advances could be less dependent on the economies of scale that are inherent in L1. Like L1, the missions that this concept set is targeted for include large-scale electricity production, management of wastes, and creation of fissile material.

The main concepts included in this concept set come from three different countries: the U.S. , Japan and the Republic of Korea (see Table 4).

Table 3. Strengths and weaknesses of the TWG 3 concept Set L1.

Strengths	Weaknesses
• Utilizes entire natural resource of fissionable material • Reduces physical and technical demands on repositories - Number of repositories - Mass of disposed waste - Volume - Decay heat - Toxicity - Confinement lifetime • Offers a plutonium utilization (Pu management) option other than repository burial • Spent fuel is either within containment or massively shielded hot cell • Separating pure Pu is not part of the reference process	• Recycling facilities would exist (remote process technology dispersed) • Degree of difficulty in reconfiguring process plants must be established
• Safety case relies on passive response • Ambient pressure primary forecloses loss-of-coolant vulnerability. • Classical ATWS events cause no fuel damage • Decay heat removal system needs no forced coolant circulation. • Long response times • Dominant phenomena well understood (large margins to boiling, single phase phenomena, well characterized feedbacks) • Source term reduced because of dilution with large sodium-coolant pool, chemical reactions with sodium, reduced diffusion/transport • Accidents leading to core damage are of exceedingly low probability, well below those of DBAs • Emergency response plans likely not required • Long system time constants and effective holdup mechanisms • Core-compaction recriticality might be precluded with planned research	• Potential for operational accidents involving sodium and inert cover gases • Positive sodium-void coefficient of reactivity • Core-compaction recriticality might not be precluded with planned research
• Innovative design features pursued to reduce cost • Passive safety, enabling reductions of safety-class designation • Minimum construction period for monolithic plants • Development costs relatively low in Japan and France, where technology base and infrastructure exists	• Capital costs historically high • Development costs high in some countries

Table 4. Concepts in Set L2: Intermediate sodium-cooled, metal-alloy fueled reactors; pyroprocess.

Concept	Known As	Size (Mwe)	Fuel	Outlet Temperature (°C)	Fuel Cycle	Country	Sponsoring Organization
M1	S-PRISM	760	U-Pu-MA-Zr metal	510	Pyroprocess	US	General Electric Co.
M5	M-JSFR	500	U-Pu-MA-Zr metal	530	Pyroprocess	Japan	JNC
M7	KALIMER	150	U-Pu-MA-Zr metal	530	Pyroprocess	Rep. Korea	KAERI
M15	AFR	~300	U-Pu-MA-Zr metal	510	Pyroprocess	U.S.	Argonne

Compared to L1, the important new features here are radically new fuel cycle technology, expected to be feasible and economic at small-scale throughput, which would allow a completely new paradigm in deployment (a fuel cycle facility serving one or a few reactors). Modularity of reactor system construction is also a prominent, though not universal feature of this concept set.

The basis for the TWG 3 evaluation of L2 is primarily the GE-sponsored concept from the U.S., S-PRISM; for which proponents point out that nearly $100 million was spent on development in the late 1980s and early 1990s. Another important recent reactor concept effort that is fundamental to the TWG 3 evaluation is the Japanese JNC concept, the Modular-type JNC Sodium-Cooled Fast Reactor, or M-JSFR. Significant technical resources have been put into this design as well. On the fuel cycle side, the basis for the evaluation is primarily Argonne's extensive experience in recent years with the pyroprocess, in a spent fuel treatment mode as opposed to full fuel recycling.

In sustainability, like L1, the L2 concept set fully enjoys the advantages of maximum resource utilization, minimum waste impacts, and reduced environmental impacts discussed for L1.

Achieving passive safety rests on inherent reactivity feedbacks including those that accompany the metal alloy fuel of L2, and the large thermal inertia of the primary coolant. These features were tested in the landmark unprotected (i.e., without reactor scram) loss of flow and unprotected loss of heat sink tests in EBR-II in 1986. It was subsequently shown by analysis that similar results would be obtained in similar larger reactors. Finally, in the 1990s it became apparent that these reactors could be rather easily designed with core conversion ratios near unity, lowering the reactivity swing during a cycle, and in turn reducing needed control rod reactivity. The classical rod runout transient was therefore also reduced in consequence from that which had been contemplated until that time.

In economics of the reactor plant, there are two lines of inquiry or development that have been pursued in recent years. The first is modularity, and with it factory fabrication of most of the reactor vessel and its internals. The second line is to capitalize on the inherent and passive characteristics of the L2 concepts to cast a significantly lower fraction of the plant's structures, systems, and components in nuclear safety-grade code-class.

Table 5 shows the strengths and weaknesses of concept set L2.

Table 5. Strengths and weaknesses of the TWG 3 concept Set L2.

Strengths	Weaknesses
• Utilizes entire natural resource of fissionable material • Reduces physical and technical demands on repositories - Number of repositories - Mass of disposed waste - Volume - Decay heat - Toxicity - Confinement lifetime • No liquid wastes • Offers a plutonium utilization (Pu management) option other than repository burial • Spent fuel is either within containment or massively shielded hot cell • Processing technology incapable of pure Pu separation	• Recycling facilities would exist (remote process technology dispersed) • Inseparability of Pu-TRU must be demonstrated quantitatively
• Safety case relies on passive response • Ambient pressure primary forecloses loss-of-coolant vulnerability. • Classical ATWS events cause no fuel damage • Decay heat removal from vessel needs no active systems • Long response times • Dominant phenomena well understood (large margins to boiling, single phase phenomena, well characterized feedbacks) • Long (fuel-coolant) thermal response time; low thermal-inertia fuel and large thermal-inertia coolant inventory • Scalability: license by test part of the licensing strategy? Based on EBR-II transient testing results • Source term reduced because of dilution with large sodium-coolant pool, chemical reactions with sodium, reduced diffusion/transport • Accidents leading to core damage are of exceeding low probability, well below those of DBAs • "Offsite emergency response plans not likely required • Long system time constants and effective holdup mechanisms	• Potential for operational accidents involving sodium and inert cover gases • Positive sodium-void coefficient of reactivity
• Modular design; factory fabrication • Passive safety, enabling reduction of safety-class designation • Reduced construction period, reduced capital at risk • Reduced operating cadre? • Compact, batch approach to fuel cycle, avoiding economies of scale	• Capital cost historically too high • Development cost is high, wherever reactor and fuel cycle demonstration is necessary

1.3 Concept Set L4: Intermediate-Sized, LBE Cooled, TH-U-Pu-MA-Zr Metal Fueled Reactors; Pyroprocess Fuel Cycle

Concept set L4 consists of designs put forward by INEEL and MIT groups. The concepts in this grouping are motivated by a near term nuclear waste management function—with reference made to flexible future missions once the primary mission is accomplished. These concepts are (relatively) immature in their degree of design development. They are noteworthy in being strongly motivated to explore innovative approaches to exploit the thermal/chemical/physical properties of LBE coolant as a way to simplify the design and reduce capital cost. Table 6 summarizes some of the important features of this concept set.

Table 6. Concepts in Set L4: Intermediate LBE cooled reactors, Th-U-Pu-MA-Zr metal (or other) fueled reactors; pyroprocess fuel cycle.

Concept	Know as	Size (Mwe)	Fuel	Outlet Temperature (°C)	Fuel Cycle	Country	Sponsoring Organization
M18	In-vessel Direct Contact Steam	419	Th-U-Pu-MA-Zr metal (or other)	545	Pyroprocess	U.S.	INEEL/MIT
M19	Burner of LWR Actinides	~400	Th-U-Pu-MA-Zr metal (or other)	540	Pyroprocess	U.S.	INEEL/MIT
M23	Minor Actinide burner	~400	Pu-MA-Zr Metal	540	Pyroprocess	U.S.	MIT/INEEL
M27	Pebble Fuel	~400	Th-U-Pu-MA-Zr metal (or other)	540	Pyroprocess	U.S.	INEEL/MIT

The L4 concepts are all based on the same basic plant layout and are all targeted to the same function—that of TRU burning in a symbiotic energy system with LWRs, and proposed as a cost effective competition to the Accelerator Driven Systems for this function. They all propose to use a totally new fuel—Th/U/Pu/MA/Zr or Pu/MA/Zr metallic alloy. Each concept features a different unique innovation which seeks to exploit the properties of LBE coolant; this cluster can be viewed as a basic concept with alternative burner options (M19/23), a concept of very different (pebble bed) core design (M27) and a concept incorporating a high pressure primary with a direct steam cycle based on bubbling steam through the primary LBE to eliminate primary pumps and steam generators (M18).

As a competitor to the ADS for the waste management function, the L4 concepts introduce thorium to increase βeff, to add to the Doppler coefficient, and to reduce burnup reactivity loss-all the things needed to eliminate need for an accelerator.

In L4, further potential payoff lies in exploitation of the relative chemical inactivity of lead or LBE coolant with air and water. This chemical inertness, relative to sodium, offers numerous potential payoffs with regard to simplifying the refueling and heat transport strategies of fast reactor design. In sodium systems these are driven by a need to separate sodium, air, and water at every stage, both in normal operations and in off-normal situations as well. Less stringent measures for such separation could potentially simplify safety strategies, which could decrease capital and operating costs.

The reactors of concept set L4 are burners or fissile self-sufficient concepts operating on a closed fuel cycle with two purposes: (1) manage the back end of a symbiotic energy system by consuming the spent fuel of thermal reactors; and (2) produce affordable electricity safely. L4 seeks lower costs, compared to the further-advanced and Russian-dominated set L5, which has been set aside in TWG 3 R&D planning. This cost improvement might be attained by introducing innovations over and above those in the Russian concepts. L4 generally has larger uncertainty than set L5 for several reasons. First the Russian industrial experience base is not available in the West and has to be regenerated (to QA standards appropriate for US-NRC licensing). Second, the Russian concepts have been under development since the early 1990s and are at a rather advanced level of refinement, whereas the L4 concepts have received only 2–3 years of work. Third, the L4 concepts seek to exploit the properties of LBE with innovations over and above what has been proposed in L5—with various mixes of options for using streaming fuel assemblies (or a pebble bed core concept with hydraulic holddown); using direct contact heat exchange in a pressurized primary; coupling to a Brayton cycle; bottom entry control rods, and others.

The strengths and weaknesses of this concept set are depicted in Table 7.

Table 7. Strengths and weaknesses of the TWG 3 concept Set L4.

Strengths	Weaknesses
• Utilizes entire natural resource of fissionable material • Reduces physical and technical demands on repositories - Number of repositories - Mass of disposed waste - Volume - Decay heat - Toxicity - Confinement lifetime • Offers a plutonium utilization (Pu management) option other than repository burial • Spent fuel is either within containment or massively shielded hot cell	• Basic recycle process flowsheets are undefined • Fuel performance unknown • Recycle facilities would exist (remote process technology dispersed) • Environmental impacts of polonium • Inseparability of Pu-TRU must be demonstrated quantitatively
• All of the strengths of L2 in Safety, plus • Negligible coolant-void reactivity • Lack of coolant reaction with air or water	• Corrosion/erosion potential of lead and LBE • Hazardous nature of Pb
• Elimination of secondary coolant system • Modular design; factory fabrication (?) • Passive safety, enabling reduction of safety-class designation • Reduced construction period	• High development cost • Seismic/structural design challenge

1.4 Concept Set L6: Small, Lead or LBE Cooled, Metal or Nitride Fueled Reactors, Cartridge Refueling; Generally Pyroprocess Fuel Cycle

The *Battery Plant Size* category contains concepts using lead and LBE coolants (see Table 8), and it spans a range of mid-term to long-term market entries. The market penetration strategy is based on economy of mass production of small turnkey plants having long refueling interval.

Table 8. Concept Set L6: Small lead or LBE cooled reactors, cartridge refueling, ~pyroprocess recycle.

Concept	Known As	Size (MWe)	Fuel	Outlet Temperature (°C)	Fuel Cycle	Country	Sponsoring Organization
M11	ENHS (LBE)	125 (thermal)	Metal or nitride	564/543	AIROX or pyroprocess	U.S.	U. California
M13	STAR-LM (LBE)	120–160	U-Pu-MA nitride		Pyroprocess	U.S.	Argonne
M17	STAR-H2 (Pb)	400 (thermal)	U-Pu-MA Nitride	780	Pyroprocess	U.S.	Argonne
M21	Integrated lead reactor (Pb)	~350	Metal or nitride	540 (and up)	Pyroprocess	Brazil	IEAv/IPEN

Also, a near-term concept is implicitly included, the Russian SVBR-75/100(M2) concept which has adapted submarine LBE and UOX fuel technology, with little technology extension, to a civilian battery plant. It is a valuable entry because it benefits from an industrial final design and cost estimate.

Mid-term LBE cooled battery concepts ENHS (M11), and STAR-LM (M13) employ fuel/clad coolant options (U/Pu/Zr alloy and U/Pu/Nitride respectively) which, when coupled with LBE, require irradiation testing. They also employ natural circulation at full power, serial factory fabrication, long refuel interval, and full service regional fuel cycle centers—all of which will require substantial development, which makes them mid-term candidates.

One long-term battery concept STAR-H2 (M17) seeks to exploit the high temperature potential of lead coolant to broaden the energy services portfolio for nuclear energy; it is similar in function therefore to the Modular M21 concept. Concept M17 proposes thermochemical water splitting for hydrogen production, while M21 proposes coupling to a chemical heat pump to produce a high temperature process heat supply for diverse industrial applications. Both seek to exploit the potential of lead coolant to achieve higher temperature for a broadened role for nuclear energy, and they will require substantial materials development R&D.

Table 9 shows the strengths and weaknesses of concept set L6.

Table 9. Strengths and weaknesses of the TWG 3 concept Set L6.

Strengths	Weaknesses

• Utilizes entire natural resource of fissionable material
• Aims for higher temperature operations in some applications
• Reduces physical and technical demands on repositories
  - Number of repositories
  - Mass of disposed waste
  - Volume
  - Decay heat
  - Toxicity
  - Confinement lifetime
• Offers a plutonium utilization (Pu management) option other than repository burial
• Regionalized fuel cycle service and markets in developing countries
• Spent fuel is either within containment or massively shielded hot cell
	• Basic recycle process flowsheets are undefined
	• N-15 recycle feasibility in nitride-fueled systems must be demonstrated
	• Environmental impacts of polonium in LBE systems
	• Inseparability of Pu-TRU must be demonstrated quantitatively

• All of the strengths of L2 in Safety, plus • Negligible coolant-void reactivity • Lack of coolant reaction with air or water	• Corrosion/erosion potential of lead and LBE • Maintenance of high core inlet temperatures in lead-cooled options
• Elimination of secondary coolant system • Modular design; factoring fabrication • Passive safety, enabling reduction of safety-class designation • Reduced construction period	• High development cost

2. OVERVIEW OF MAJOR TECHNOLOGY GAPS

2.1 Concept Sets L1/L2—Major Potential Payoffs and R&D Needed

The L1/L2 sodium-cooled fast reactor concept sets incorporate significant recent innovations in the fuel cycle that will require substantial R&D during the viability R&D campaign. These innovations center around the adoption of the *dirty fuel/ clean waste fuel cycle strate*gy wherein all transuranics are recovered in a commixed stream, and recycled to the reactor for total consumption by fissioning in the fast neutron flux.

The fuel cycle based on the dirty fuel/clean waste strategy takes on special relevance for the sodium-cooled fast reactor concept sets because of its unique capability and mission in Gen IV for regulating the actinide flows for the global nuclear energy system as a whole. Sodium-cooled fast reactors – unique among all reactor types – when coupled to a closed fuel cycle can perform either as net burners of transuranics or as net breeders of transuranics, requiring only a change in core reload pattern. Thus, when deployed as a segment of a symbiotic nuclear energy system, the sodium-cooled fast reactors can be used to moderate the overall ebb and flow of actinide inventories—burning as needed when LWR spent fuel accumulates and later creating fissile material when and if it runs short due to diminishing stocks of low-cost uranium ore. The payoffs from using concept sets L1/L2 as the actinide manager for Gen IV are: (1) maximizing resource use; (2) minimizing waste production; and (3) placing intrinsic safeguards features on the Gen IV fuel cycle (no separated plutonium, only trace amounts of fissile material sent to the repository, and materials meeting the spent fuel standard of self protection at every link of the fuel cycle chain).

In the burning mode, because of their high energy neutron spectrum, the steady-state isotopic mass distribution in fast reactors is not skewed to the heavy (and toxic) isotopes as would be the case with softer-spectrum systems. Thus, when sodium-cooled fast reactors are used as waste burners, consuming the discharge from other reactors in the system, the trace losses of transuranics sent to the repository contain few minor actinides. This is an non sequiter advantage in minimizing the toxicity and decay heat of the waste stream going to the repository.

In the fissile-creation mode, because of high power density—unattainable with gas and lead-based fast reactors—the sodium-cooled systems can achieve short doubling time. This makes the sodium-cooled reactors an essential and irreplaceable component of sustainable future nuclear energy systems. At a future time, all reactors (fast and thermal) in the system will depend for their fuel supply on the breeding of U-238 into transuranics in the sodium cooled fast reactors. As shown in the Gen IV FCCG scenarios, only sodium-cooled reactors with their short doubling time are capable of keeping up with demand in a growing system.

The dirty fuel/clean waste strategy raises technology issues and knowledge gaps to be addressed by viability R&D. These include:

- Effect of minor actinide and fission product carryover on fabrication technology and on fuel performance. Development of cost effective remote fabrication
- Development of cost effective recycle chemistry which minimizes trace losses of transuranics to waste
- Development of cost effective waste forms.

A very substantial technology base exists already for sodium-cooled fast reactors. Thus, for the power plant itself, the Gen IV Viability R&D phase of the program will be directed especially to a small number of remaining technology gaps. These include, broadly speaking:

- Capital cost reduction measures
- Assurance of passive safety
- In-service inspection and repair technologies.

For its traditional electricity mission of fissile material and electricity production, the technology is far along in development. But for the dirty fuel clean waste mission, with its many new benefits, significant technology development remains, particularly in the fuel cycle.

2.1.1 Fuel Cycle R&D

With the pyroprocess fuel cycle technology it will be essential to conduct plutonium and minor actinide extraction experiments from electrorefiners at a much larger scale than has been done until now (~50 g plutonium). Significant work on electrorefiner salt cleanup and high-level waste form production needs to be done in order to achieve the very high actinide recoveries (~99.9%) that are the objective of the process. It is important to develop any secondary waste stream treatment that may become necessary to achieve this recovery goal. Also, it is necessary to complete certification of the two high-level waste forms (metal and ceramic) for repository disposal.

Safeguards implications of an experimental pyroprocess facility operated in a fuel recycle mode (building on the current experience of pyroprocessing operating in a waste management or spent fuel treatment mode) must be investigated.

With advanced aqueous processing, viability R&D work remains to be done to demonstrate the high actinide recoveries (99.9%) (and also the long-lived fission product recovery and disposition when that is a central part of the process strategy). For aqueous facilities especially, the proliferation resistance features need demonstration. For countries that have active aqueous processing programs, these developments can be carried out in adapted existing facilities.

Since ceramic pellet fabrication by the traditional approaches is not likely to be successful with the radioactive fuels that result from the advanced aqueous process, it is important to demonstrate the proposed remote fabrication processes for ceramic fuels, whether the process is simplified pellet fabrication or one of the particle compaction approaches.

2.1.2 Fuel Performance R&D

Among the more significant technology gaps for fast reactor systems using recycled fuel is a need for performance data and transient safety testing of fuel that has been recycled using prototypic processes, including refabrication using prototypic remote approaches. This fuel will have concentrations of uranium, plutonium, and minor actinides and carryover fission products. The fuel morphology will derive from the specific recycle/refabrication technologies being developed in Gen IV. This dependency is true for the oxide and metal fuels of concept sets L1/L2, which already have a partial data base, as well as for the fuels of L6/L4 (which don't). Therefore, there is a large requirement for facilities, in particular for remotely operated fabrication facilities, as the test programs are as much a test of fabrication as of fuel performance. (But the throughputs do not have to be large if a lead test assembly (LTA) strategy is adopted. This requirement can either be considered a fuels issue or a fuel cycle issue.)

In addition, for high-burnup metal fuel, transient testing to damage conditions may be needed for establishing the safety case for these systems.

2.1.3 Reactor R&D

Innovations for the L1/L2 concept set include various means to reduce capital cost. Both economy of scale (L1 concepts) and economy of modular factory fabrication and just-in-time capacity additions (L2 concepts) are proposed—i.e., strategies optimized for the financial parameters that may exist in regulated or deregulated markets.

For L1 concepts, the design innovations include simplification based on reducing the number of loops and simplifying and increasing the size of components. Here the availability of qualified advanced materials (for example 12Cr-1Mo) is considered a technology gap requiring viability R&D.

The L1/L2 concept set benefits from innovative balance of plant simplifications—including the use of passive safety design approaches to reduce or eliminate the assignment of safety functions to balance of plant equipment. Additionally, simplification of the BOP energy conversion by switching from the Rankine steam cycle to innovative Brayton cycle equipment holds promise. (The R&D needed for a supercritical CO_2 Brayton cycle operating at sodium outlet temperature is described with the L6/L4 concept set, and is crosscutting with the L1/L2 set as well.)

In O&M technology, a gap exists with sodium-cooled reactors (and with lead-cooled systems too) in the area of in-service inspection and repair. The viability R&D involves development of under-sodium viewing and/or ultrasonic testing in sodium.

In physics, a technology gap exists because of the recycling of all minor actinides. Basic nuclear data enhancements are needed for at least certain of these isotopes.

In reactor safety the gaps center around three general areas: basic properties; assurance of passive safety response, including the modeling and validation of the models through experimentation; and the technology for evaluation of "bounding events," i.e., analysis of less likely events but with potentially higher consequence. The R&D plan to fill the remaining knowledge gaps is organized around the major off-normal events or potentials; transient overpower and undercooling; assurance of adequate passive decay heat removal; reactor shutdown assurance; cladding integrity; and containment integrity. TWG 3 believes that with development to remove the remaining technology gaps, it can be shown that all L1/L2 systems can accommodate all design basis accidents and most beyond design basis accidents with passively safe response and no core damage.

It is generally believed that even though the design basis will not include events leading to core disruption, present day licensing practice and the licensing history of sodium-cooled reactors will likely lead to a requirement to include analysis of some extremely low-probability "bounding" accident sequence that would lead to core disruption. The U.S. members of TWG 3 and their technical support cadre believe that for L2 (metal fueled) systems, even these extremely unlikely events can be accommodated passively with no possibility of core compaction and re-criticality. The Japanese members of TWG 3 and their technical support cadre believe that, at least for the oxide-fueled systems of L1, elimination of re-criticality can be assured with design modifications and further R&D, and the JNC "recriticality free" program is included under "other special topics" in the R&D plan.

Recent events increase the attention that must be paid to physical security for future reactors and fuel cycle facilities. Models are needed to test different conceptual approaches, and experiments may be needed to validate the models.

2.2 L6/L4 Lead or LBE-cooled Reactors

The technology gaps for these systems first center around the objectives of the science-based program: settle coolant compatibility issues, and aim for higher operating temperatures. Coolant/clad/fuel performance R&D is necessary, initially at temperatures of ~550°C, with the objective of higher temperatures (800°C or higher, perhaps) as developments proceed.

Identification of the most promising fuel candidates is needed. Depending on the specific choice of fuel (nitride, metal, oxide), steady-state irradiation performance and transient fuel testing will be required, as noted above.

In the fuel cycle, if higher temperature operations lead to nitride fuel, irrespective of the fuel cycle chosen, it may be an economic imperative to recover and recycle the nitrogen, fully enriched in N-15, in order to avoid large C-14 production. If the pyroprocess is to be used, as suggested by a number of concept principal investigators, the use of nitride fuel involves all of the technology gaps of the pyroprocess discussed in the previous section. In addition there are further important technology gaps: the capture of the N-15 in the head-end steps, the reconversion of the metal product to nitride; and the remote fabrication of the nitride ceramic fuel.

Reactor safety technology gaps bear resemblance to those of sodium-cooled systems for events within the design basis. Local flow blockage and response, and passive feedback assurance appear as the most important. Not surprisingly, little is known at present on how the beyond-design-basis events might be treated in lead-cooled systems, but the significant reduction in coolant void coefficient will help greatly.

In reactor technology, the main technology gap is structural materials for primary system components. In-service inspection and repair issues are the same as for the sodium-cooled systems.

In physics, basic nuclear data enhancements are needed for lead, bismuth, and certain of the minor actinides.

3. R&D PLAN FOR CONCEPT SETS L1/L2

In this major section of this R&D Scope Report, the R&D plan is presented for development of sodium-cooled reactors and fuel cycles. As noted before, great emphasis in Gen IV is being placed on the R&D necessary to close the fuel cycle. TWG 3 put major emphasis on these technology issues as noted before. Most importantly, the liquid metal-cooled reactors will operate for a long time in concert globally with large numbers of thermal reactors. The fuel cycle technologies must be capable of accommodating this reality.

While the emphasis is on the fuel cycle technologies, attention must also be paid to the reactor technologies. This is particularly important in assuring passive safety, in aspects of reactor design aimed at cost reduction, and in nuclear security.

3.1 Fuel Cycle

Three fuel cycle processing technology options will be developed, with emphasis on the first two: (1) advanced aqueous processing (Section 3.1.1), and (2) pyroprocessing (Section 3.1.2). A third process option is also developed, termed “other dry processes” (Section 3.1.3). In each of these processes pure plutonium is never separated, and it is always accompanied by uranium and minor actinides, and at least trace amounts of fission products. This implies that fuel refabrication technology must be done remotely, in hot cells. Equipment must be operated, maintained and repaired using remote means. This is given special emphasis in the R&D plan.

The specific advanced aqueous process discussed below (and the dry process discussed in Section 3.1.3) is from Japan, where the largest amount of work in these areas is now in progress. It is recognized that advanced aqueous processing is being discussed or in development elsewhere, too. Regardless of its specifics the approach taken to the R&D is thought to be at least broadly similar to what would be to undertaken elsewhere.

3.1.1 Advanced Aqueous Processing and Remote Fabrication

The combination of advanced aqueous reprocessing and an advanced pelletizing method is considered a suitable option for MOX fuel cycle closure of the L1 concept set. The discussion in this section is the approach taken in Japan. Figure 1 and Figure 2 show the schematic flow diagrams of these reprocessing and fuel fabrication systems, respectively.

The advanced aqueous reprocessing basically consists of “simplified PUREX”, with the addition of a crystallization process and a MA recovery process. The purification process of U and Pu in the conventional PUREX is eliminated, resulting in co-recovery of U/Pu/Np with relatively low decontamination factor (DF) for recycle use, which is favorable for proliferation-resistance. The crystallization process is adopted for the recovery of excess U before co-recovery of U/Pu/Np (In the U.S. the UREX process is under investigation, for the same reason of removing bulk uranium in a head-end step). An MA recovery process is also adopted to collect Am and Cm. The Pu content in the solution fed into fabrication is pre-adjusted in the advanced aqueous reprocessing. The pelletizing process is simplified by eliminating the MOX powder handling processes from mixing to granulation in the conventional MOX pelletizing process. In this combined fuel cycle system, greater than 99% of U/TRU is expected to be recycled, and the DF of the reprocessed product is expected to be greater than 100. Processing LWR spent fuel by this means is also considered as an option, so that initial core loads for the fast reactors could be provided.

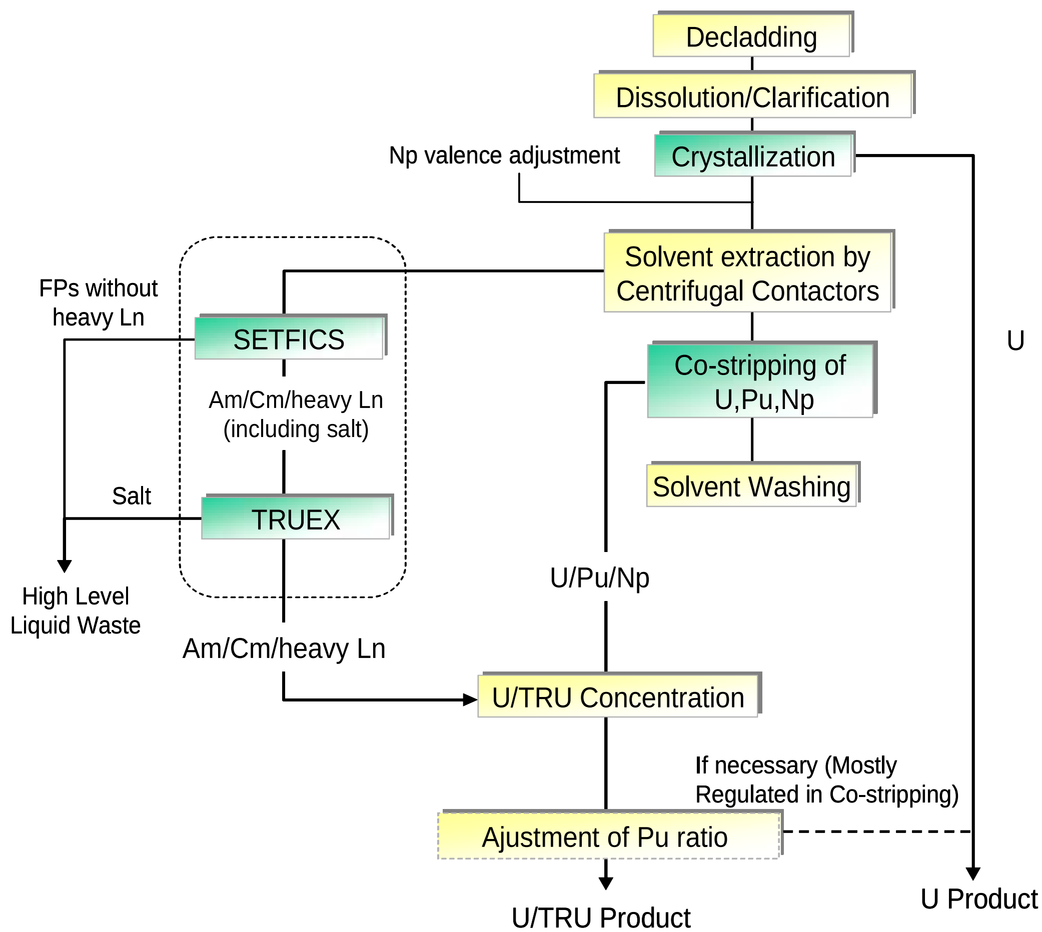

Figure 1. Schematic flow diagram of advanced aqueous reprocessing (simplified PUREX process with crystallization and MA recovery).

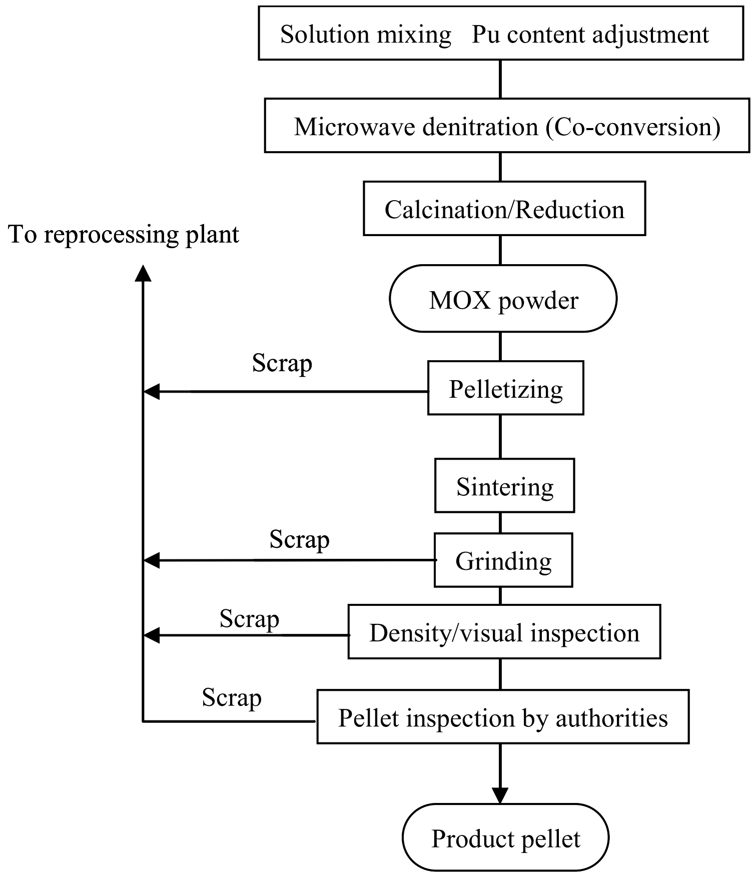

Figure 2. Schematic flow diagram of the advanced pelletizing process.

With regard to this system, only a few viability R&D issues are thought to exist, since the process technology builds heavily on prior fuel cycle technology for LWRs and fast reactors. Therefore, proponents claim that this fuel cycle system can be advanced to the demonstration stage rather quickly. Priority is placed on the economic competitiveness of the process, and on process development that reduces the environmental impact. Studies in Japan on an advanced aqueous reprocessing system for the fast reactor cycle show the economical plant capacity for processing fast reactor spent fuels, with one line of equipment will be around 200 MTHM/y. Research and development in Japan for the facility and equipment will therefore be oriented toward this 200 MTHM/y plant.

Before completing the design for an oxide facility, additional laboratory-scale tests, tests with irradiated materials, and an integrated demonstration are needed. Facilities exist in Japan for both laboratory-scale tests and initial tests with irradiated materials. For an integrated demonstration to obtain data to support the design of a commercial plant, a new facility is needed.

Consistent with RIT guidance, the R&D items for each process step (or set of steps) are classified as follows. These classifications will be used throughout the text and in Appendix A:

> *Viability R&D* is development of new technology. It must be accomplished to confirm technological feasibility. If the expected results are not obtained, a substitute system or process must be developed.

Performance R&D is the improvement of conventional technology and development of systems or equipment after the viability R&D is done. This is needed to confirm commercial application of the system. The demonstration of commercialization technologies requires engineering-scale tests.

The R&D items for this system are as follows:

3.1.1.1 Head End Processing and Fuel Preparation.

a. Disassembling and shearing

In order to improve economic competitiveness, R&D on disassembling and shearing is proceeding, to downsize the process using a laser cutting device and a new shearing process. The integrated disassembly/shearing system using a laser is expected to achieve 50% downsizing of disassembly/shearing equipment.

a. Decladding

The decladding and fuel powdering process is not considered essential to the advanced aqueous reprocessing system. However, the increase in dissolution rate by dissolving powdered spent fuel is expected to minimize dissolver complexity and/or to lower temperature, and to obtain a high HM concentrated dissolution suitable for the next-step crystallization process. To get powdered spent fuel, the mechanical decladding system such as a shredder has been examined. Powdering performance of spent fuel and separation efficiency of powdered claddings from spent fuel must be studied for this decladding system.

3.1.1.2 Advanced Aqueous Process.

a. Crystallization process

A crystallization process is adopted for pretreatment in the simplified PUREX process in order to reduce the amount of solution and solvent in the follow-on extraction process, and to downsize the processes downstream. In the crystallization process, the dissolved solution is cooled down and excess U is precipitated as a crystal of uranyl nitrate hexahydrate (UNH) according to the solubility at the lower temperature. It is expected that the decontamination factors for fission products in the UNH product are approximately 100 (from a stimulated dissolver solution test). It is necessary to control and optimize the UNH crystallization conditions with little Pu content, both because it affects plutonium and minor actinide recovery fraction, and because of the criticality safety of the equipment that handles crystallized UNH. The results of small-scale hot tests indicate that the co-crystallization behavior of Pu with UHN depends on the valence of Pu. Therefore, the crystallization behavior of actinides should be studied in addition to obtain the UNH yield and its DFs for TRU and fission products in small-scale hot tests. Further the optimization of the UNH crystallization will be demonstrated in an engineering-scale uranium test.

b. U/Pu/Np co-recovery

In order to co-recover Np with U and Pu, a complete oxidation to the valence state VI is required. The valence of Np can be easily adjusted by controlling the solution temperature. Np recovery in a mixed U,Pu product solution has been demonstrated with DFs over 1000 in small-scale hot tests and the first cycle of the Tokai Reprocessing Plant.

A centrifugal contactor is expected to increase the extraction rate in this process. The salt-free process is required (see immediately below), which results in the reduction of the low-level waste. It should be studied to optimize U/Pu/Np recovery condition and the DF for fission products.

c. MA recovery

The combination of SETFICS and TRUEX (see Figure 1) process using TBP and CMPO is applied to this system as the MA (Am,Cm) recovery process. Small-scale hot tests were implemented to investigate the separation efficiency of MA from lanthanides. In the TRUEX process, sodium nitrate is used which causes an increase of HLW. . It was confirmed with cold tests that a salt-free reagent such as hydroxylamine nitrate (HAN) was applicable in this process. However, a salt-free MA recovery process should be studied in small-scale hot tests.

d. Development of dissolver equipment

Feasibility of an annular continuous dissolver has been verified in engineering-scale cold tests and small-scale hot tests. Engineering-scale cold tests and tests with uranium are required for evaluation of the endurance and reliability of this component. Further, the calculated results show that the overall dissolution rate of powdered fuel is expected to be 10 times that of chopped fuel pieces. However, there are little experimental data on the dissolution of powdered fuel. Therefore, small-scale hot tests are required for understanding the dissolution performance.

e. Crystallization equipment

For the crystallization process, criticality safety plays an important role in the design of the crystallizing device. Moreover, crystallization performance of uranium, separation efficiency of solids and liquids, transportation of UNH crystals, and the conversion and storage method of the uranium product should be studied in engineering-scale cold tests and tests with uranium.

f. Centrifugal contactors

The feasibility of using centrifugal contactors in the fast reactor fuel cycle has been verified at small-scale (1–4 stages) in cold tests and those with uranium. As a development item, the endurance and reliability of the drive unit of the centrifugal contactors, system operability, and co-decontamination performance of single-cycle solvent extraction should be studied for the development of centrifugal contactors in engineering scale uranium tests.

g. New processing methods of advanced aqueous processing

To further improve economic competitiveness and to reduce the volume of radioactive wastes, innovative processes such as alternative direct extraction methods with supercritical CO_2/TBP/HNO_3 or TBP/HNO_3 under normal pressure and temperature, amine extraction method, and an ion exchange method, might be studied.

3.1.1.3 Pellet Fuel Fabrication. MOX pellet fabrication technology based on glovebox confinement has been verified in the highly decontaminated plutonium recycling system in the commercial LWR and fast reactor fuel cycle. In the fast reactor fuel cycle, with low-decontaminated MA bearing fuel, however, major R&D is required to confirm the operability of the fabrication system in a hot cell facility.

a. Denitration/conversion (Turntable system)

Viability R&D is associated with feasibility tests of the denitration/conversion system, in which each of the reactions of denitration, calcination and reduction take place in the same container. A bench scale apparatus (a glove box) will be manufactured (high decontaminated fuel is to be used in the testing), and it must be confirmed that every reaction proceeds efficiently.

Performance R&D involves provision of an engineering scale apparatus in which denitration/conversion system testing can be accomplished including features such as nitrate solution feeding, denitration, powdering, calcination, reduction and discharge of MOX powder. The operational condition for each process step will be ascertained, and the process data collected. R&D for automating the process and for remote maintenance will be done.

Performance R&D also includes provision of an in-cell apparatus (turntable system) which has 6 stages, with about 2 kgHM/batch throughput. This will establish the economic viability. MOX powder for irradiation testing will be processed, and process data for design and construction of a commercial plant will be obtained.

b. Fabricability of MA-bearing fuel

Assuming that minor actinides will be present in future fast reactor fuel, fabricability of MA- bearing pellet fuel (such as sinterability) must be established.

Devices for each of the in-cell process steps of pressing, sintering, grinding and pellet inspection will be provided, based on the existing R&D results. Process data for design and construction of a commercial plant will be obtained through fabricating fuel for irradiation testing.

The apparatus will be examined for its system operability and maintainability in a hot cell facility. Fabrication cost will also be evaluated.

c. ODS cladding welding/welding inspection

Performance R&D will provide a welding/welding inspection apparatus by pressurized resistance welding and ultrasonic inspection methods. Welding/inspection of fuel pins for irradiation testing will be carried out. Process data such as the stability of welding, and welding/inspection speed will be obtained.

d. Quality control/process control

The quality control/process control systems associated with lower decontamination-factor processes and recycle of minor actinides must be developed. Experimental equipment will be provided and process data will be collected in this R&D element.

e. Accountability

R&D is required for verification of the analysis method and analysis system, and for study of the safeguards concept for pellet fabrication processes in a large hot cell environment.

f. Remote maintenance

Because the fuel fabrication equipment will be operated in a hot cell, complete remote maintenance is required for all equipment. Performance R&D items concerning remote maintenance technologies will be addressed.

g. Alternative sphere-packed fuel fabrication option

Sphere-packed fuel fabrication based on the gelation process applied to an aqueous processing system is expected to realize dust-free fabrication, which in turn will minimize TRU migration to the waste. This system must also be remotely operable and maintainable. In the case of low-decontamination and minor-actinide-bearing fuel, it will be necessary to show the applicability of gelation to multi-component systems.

3.1.1.4 Waste Management.

a. High-loading vitrification process of HLW

Fission product content in HLW is about 10%. If Mo, Sr and Cs are removed, fission product content of HLW can be increased to about 30%. If enough decontamination is achieved in a Mo recovery process, Mo can be disposed as TRU waste. Sr and Cs can be disposed as HLW, in the high-loading form mentioned, after sufficient interim storage.

b. TRU waste

The largest source of TRU waste generation in conventional aqueous processing comes from sodium reagents. They are used in solvent washing and off-gas treatment. Replacing the sodium reagent with a salt free reagent would reduce TRU waste generation appreciably. Small-scale cold and hot tests are required.

The second largest source of TRU waste generation is spent equipment. Improved decontamination technology can reduce wastes to be disposed to a lower waste category. This would enable the reduction of total disposal cost.

3.1.1.5 Process Control and Accountability. The main example of required R&D is verification of the analysis method and analysis system, and study of the safeguard concept for advanced non-aqueous processes for possible application.

3.1.2 Pyroprocess and Remote Fabrication

The following section defines areas in the pyroprocessing fuel cycle for which technology gaps exist and research and development (R&D) activities are necessary to close those gaps. The required R&D may be divided into two parts as defined before: (1) Viability R&D, including basic research and proof of phenomena, needed to bring a concept to the stage of conceptual design of a prototype; and (2) Performance R&D, including technology development and proof of practicality, needed to carry through the first-of-a-kind engineering stage.

The pyroprocess fuel cycle as discussed in this section can be divided into two categories, the cycle for oxide fuel and the cycle for metal fuel. The recovered material from the oxide cycle serves as feed material for the metal cycle. There are many common characteristics and process steps in these cycles. This section of the report is divided into the key process steps, and the needed R&D is noted for each step. Different requirements for treatment of oxide and metal fuels are noted. Only the first process step, front-end conversion, is unique to oxide fuels.

Much of the R&D for the oxide cycle falls into the viability category. The ability to reduce spent oxide fuel to metal has been demonstrated but not to the point that a prototype conceptual design can be completed. For the pyroprocessing fuel cycle of metal fuels, significant R&D falls into the performance category. The metal fuel pyroprocess technology is based on work performed as part of the treatment of EBR-II fuel in the Fuel Conditioning Facility (FCF) at Argonne National Laboratory-West from 1994 onward. Although the key process steps like electrorefining and cathode processing are at the performance R&D level, viability R&D is still needed for the metal cycle, for example to demonstrate the recovery of actinides beyond uranium, and to resolve materials of construction issues. Work in these areas is also required for oxide fuels.

The goal of this R&D plan is to complete the design of commercial pyroprocessing facilities for both oxide and metal fuels. Under this plan, both tasks would be finished in 2015. The work breakdown structure for this plan and schedule follows the same outline as the report that is generally in the order of the unit operations.

Before completing the design for an oxide facility, additional laboratory-scale tests, tests with irradiated materials, and an integrated demonstration are needed. Facilities already exist for both laboratory-scale tests and initial tests with irradiated materials. For an integrated demonstration on a scale needed to obtain data to support the design of a commercial plant, new facilities are needed. The integrated demonstration would need to be on the order of 100 MTHM per year to support the design of a commercial facility to process more than 1000 MTHM per year.

Much of the on-going program in FCF can support the design of a commercial pyroprocessing facility for metal fuel. The present throughput in FCF is ~2.2 MTHM per year, but could be increased to ~5 MTHM with process improvements. The 5 MTHM rate is sufficient for demonstrating the metal fuel cycle for concepts in which the pyroprocessing facility is co-located with the fast reactor. The activities in FCF at present do not include the R&D needed to support the recovery of actinides beyond uranium, but FCF and the laboratory facilities at ANL could be used for this viability testing and for an integrated demonstration of the metal fuel cycle including recovery of all actinides and remote fuel fabrication.

3.1.2.1 Front End Conversion. The pyroprocess fuel cycle can make use of fissile material from both oxide and metallic fuels. For the recovery of actinides from oxide fuels for production of metallic fuels, the actinide oxides must be first reduced to the metallic state. Various flowsheets have been developed for accomplishing this task by chemical and electrochemical means. The feasibility of this concept has been demonstrated, but only at the laboratory-scale. The front end conversion of oxides to metals is the critical step in the oxide cycle and is given the highest priority in the R&D program.

a. Oxide Reduction

Viability R&D is needed to demonstrate the extent of reduction to metal that is achievable for both the actinides and rare earth fission products; to determine the properties of the fuel and the degree of cladding removal needed for complete reduction; and to demonstrate the ability to scale the process to higher production rates. Testing of this process with irradiated fuel is then needed. Limited demonstrations of this technology on an engineering-scale with irradiated materials could be performed in existing facilities. An integrated demonstration of the technology to assess throughput and costs for a production facility would require a new facility.

3.1.2.2 *Fuel Preparation.* Fuel preparation includes disassembly of fuel assemblies and shearing or chopping of fuel elements so that the fuel matrix can be exposed for the electrorefining. The disassembly and shearing of both metallic driver and blanket fuel are routinely performed in FCF.

a. Shearing

Techniques for shearing and shredding oxide fuel for reduction need to be developed and demonstrated. The degree to which the cladding is removed and the required size of the fuel particles need to be established from laboratory-scale tests of the reduction process. This size reduction of oxide fuel for treatment is required for both the pyroprocess and advanced aqueous cycles.

For metal fuel, process improvements may be needed to increase production beyond the 5 MTHM per year rate that is planned, but this rate is sufficient for most perceived metal-fueled reactor concepts.

3.1.2.3 *Electrorefining.* Electrorefining is the critical purification step in the pyroprocessing fuel cycle. In this step, actinides are separated from the bulk of the fission products. Electrorefining of metallic fuels has been demonstrated extensively as part of the treatment of EBR-II fuel.

a. Electrorefining of Reduced Oxide

As with oxide reduction, the demonstration of this step with irradiated materials is needed. A key aspect that needs to be demonstrated is that the reduced actinide metals do not have sufficient oxide impurities to negatively impact electrorefining. Specifically the carryover of actinides and rare earths that are not reduced completely during the front end conversion can impact actinide recovery during electrorefining. Because of potential differences in the base salts for electrorefining and oxide reduction, optimization of salt carryover from the reduction step to electrorefining is needed to minimize waste volume. Further R&D is also needed to determine if a different electrorefiner from that used in the metal fuel cycle is needed or beneficial for this step.

b. High-Throughput Electrorefining

In the pyroprocess fuel cycle for metal fuels and for the reduced actinides from oxide fuels, large quantities of metals must be electrorefined to separate most of the uranium from the bulk of the fission products and transuranics. For the metal fuel cycle, the electrorefining rate being implemented in FCF is very close to that required for a production plant co-located with a fast reactor.

This rate will need to be increased further to meet the requirements of the pyroprocess fuel cycle for treating oxide fuels. Improvements are also being sought in the decontamination of noble metal fission products from the recovered uranium and in increasing the purity of the recovered uranium. The required impurity levels will be driven by the fuel specification.

c. Actinide Recovery

The work performed with EBR-II fuel has not focused on the recovery of actinides other than uranium. Complete actinide recovery methods were developed and tested on an engineering scale. A demonstration with irradiated fuel needs to be performed. Additionally, for the process that was developed, improvements to increase throughput may be needed. For the pyroprocess fuel cycle, R&D to support actinide recovery is second only in importance to the oxide reduction tasks.

d. Actinide Drawdown

For a complete pyroprocess fuel cycle in which all actinides are recovered and their content in waste streams are minimized, methods to lower their concentrations in electrorefiner salts are needed before the salts are treated as wastes. For actinide drawdown, the actinide chlorides are chemically reduced and separated from the salt phase. Techniques demonstrated for accomplishing this task include both batch processing and continuous processing using high-temperature centrifugal contactors. A demonstration of a drawdown process with salt from the processing of irradiated fuel is needed as well as an assessment of its impact on overall throughput. The earlier demonstrated processes might not meet the required process throughputs, and development of improved techniques may be needed.

3.1.2.4 Fuel Fabrication. The fabrication of metallic fuel from feedstock materials was performed routinely for decades. A number of aspects of fabricating fuel from recycled materials was also demonstrated during parts of the operating history of EBR-II.

a. Salt Removal from Cathodes

The actinides are recovered from electrorefiners in various forms of cathodes, all of which have adhering salt. This salt needs to be separated from the recovered metal. The separation is accomplished using a distillation process employing elevated temperatures and vacuum. It was demonstrated extensively during the treatment of EBR-II fuel. Improvements are needed to further increase the process throughput beyond that planned for EBR-II fuel treatment.

b. Fuel Casting

Fuel for EBR-II was produced over the last few decades by injection casting. This process was demonstrated extensively. The maximum fuel slug length produced by injection casting is limited. An assessment is needed to determine if longer fuel elements can be produced by combining slugs or if a new technique is required. The use of multiple slugs in a fuel element was used in EBR-II, Fermi, and FFTF. Additionally, the process throughput needs to be assessed to determine if advancements are needed to meet increased production throughputs. Injection casting for EBR-II employed quartz tubes to produce fuel slugs. The tubes were broken after casting to recover the slugs. New injection casting molds or improved techniques are needed to minimize the loss of fissile material in the quartz waste stream.

c. Crucible Materials

Both casting and salt distillation employ high-temperature operations with molten actinide metals and salts. Under these conditions interactions occur between the melts and crucibles resulting in the formation of dross streams. Additionally the crucibles or coatings on the crucibles are not always reusable. Material testing is needed to develop reusable melt crucibles that minimize the formation of dross. Reusable crucibles are also needed to increase process throughput.

d. Fuel Fabrication

A large quantity of metal fuel was fabricated for EBR-II. Much of it was fabricated in gloveboxes using feedstock materials. In the 1960s, significant quantities were fabricated remotely in FCF as part of the demonstration of melt-refining for recycling metal fuel. The remote fabrication of fuel was part of the planned demonstration of the Integral Fast Reactor Program that was terminated in 1994. The equipment to perform these operations was built, installed, and qualified for remote operations, but it was never used to produce recycled fuel. The demonstration of this technology including remote welding of fuel caps on fuel elements and remote quality assurance inspections is needed.

3.1.2.5 Metal HLW. Two high-level wastes are produced as part of the pyroprocess fuel cycle. The first is the metal waste. It consists of the cladding materials along with those fission products that are not oxidized to chlorides during electrorefining. It includes technetium. Much of the repository qualification work for the metal waste from the metal fuel cycle has been performed as part of the treatment of EBR-II fuel. Repository corrosion models are being developed and characterization and consistency tests are being assessed.

a. Waste Qualification

On-going waste qualification activities need to be completed and all the supporting data qualified. The metal waste from the reduction of oxide fuels may have a different composition because oxide fuel cladding is zirconium while fast reactor metal fuels use stainless steels. If a modified high zirconium content waste form is chosen, additional characterization work is required to reach the same qualification level as that of the stainless steel-based metal waste.

b. Production Process Qualification

The production of the stainless steel metal waste has been demonstrated using irradiated materials. Presently production-scale equipment is being developed and tested. These activities need to be completed. The production process for the zirconium-based metal waste will be very similar. It still needs to be demonstrated with irradiated materials.

3.1.2.6 Ceramic HLW. The second high-level waste from the pyroprocess fuel cycle is the ceramic waste. The ceramic waste is a zeolite-based waste form that stabilizes the fission products that form salts in the electrorefiner. As with the metal waste, much of the repository qualification work has been performed as part of the treatment of EBR-II fuel. Repository corrosion models are being developed and characterization and consistency tests are being implemented.

a. Waste Qualification

On-going waste qualification activities need to be completed and all the supporting data qualified. If changes are made in the electrorefiner system for oxide reduction, some modifications may occur in the ceramic waste. If the modifications are significant then some additional qualification work may be required.

b. Production Process Qualification

The production of the ceramic waste form has been demonstrated using irradiated materials. Presently production-scale equipment is being developed and tested. These activities need to be completed. Again, if changes are made in the electrorefiner system for oxide reduction, some modifications may occur in the production process for the ceramic waste.

c. Ion Exchange Process

For the process presently employed, the ceramic waste is produced by what is termed the throwaway option. In the throwaway option, fuel is treated until either a fission product, sodium, or transuranic concentration limit is reached in the electrorefiner salt. When that limit is reached, all of the salt is disposed in the ceramic waste. An ion exchange process could instead be used to remove the limiting elements from the salt, and the salt could be recycled. This process has the potential to significantly reduce the waste volume. The process has been tested on a laboratory-scale. Additional thermodynamic and kinetic data are needed before it is scaled further. A demonstration with salt from the processing of irradiated fuel will also need to be performed.

3.1.2.7 Process Control and Accountability. The treatment of irradiated EBR-II fuel has occurred in the FCF since 1996. Process models and material control and accountability systems were developed and implemented for these operations to meet the requirements of the U.S. Department of Energy. Advancements in the system may be required for large-scale application of this technology.

a. Process Modeling

Process models have been developed for most of the batch operations in the pyroprocess fuel cycle for metallic fuel. Further validation of the models are required. Development of models for the oxide reduction operations are needed.

b. Accountability Development

An accountability system was implemented for the treatment of EBR-II fuel. The system meets the requirements of the U.S. Department of Energy. Process improvements to increase turnaround on sample results would be needed as processing rates are increased. Additionally, an assessment of additional international requirements will be needed.

c. International Fuel Cycle Transparency

An assessment of the pyroprocess fuel cycle is needed that addresses the transparency of the system and improvements that can be made to the materials control and accountability system.

d. Nondestructive Analysis

The development of NDA techniques to assay the various process streams would be beneficial to support the required safeguards and high process throughputs. It would also be beneficial in assessing fuel inputs, especially for the oxide fuel cycle in which satisfactory compositional data may not be available.

3.1.3 Other Dry Processes and Vibropac Fuel Fabrication Option

An oxide electrowinning process (as the other dry process considered in TWG 3, besides the pyroprocess), and fuel fabrication by vibratory compaction (termed “vibropac”, which uses a crushing technique) is considered as an option for L1, including LWR recovery. Figures 3 and 4 show the schematic flow diagrams of these reprocessing and fuel fabrication systems, respectively.

The key technologies of oxide electrowinning processing are based on technologies developed by RIAR (Research Institute of Atomic Reactors) in Russia. In Japan, a modified oxide electrowinning processing based on the RIAR method has been studied. The modifications to the RIAR method include the adoption of a simultaneous fuel dissolution and UO_2 deposition process to increase the processing speed and reduce the required amount of chlorine. A MOX co-deposition process has been adopted in place of the Pu precipitation process to recover U and Pu simultaneously. Further, an MA electrowinning process has been added to improve the MA recovery rate.

With regard to this system, confirmation of the technological feasibility of the process including MOX co-deposition is needed. Moreover, verification of MOX co-deposition is one of the important

R&D activities. The feasibility of these core technologies of the oxide electrowinning process will be assessed through small-scale plutonium tests.

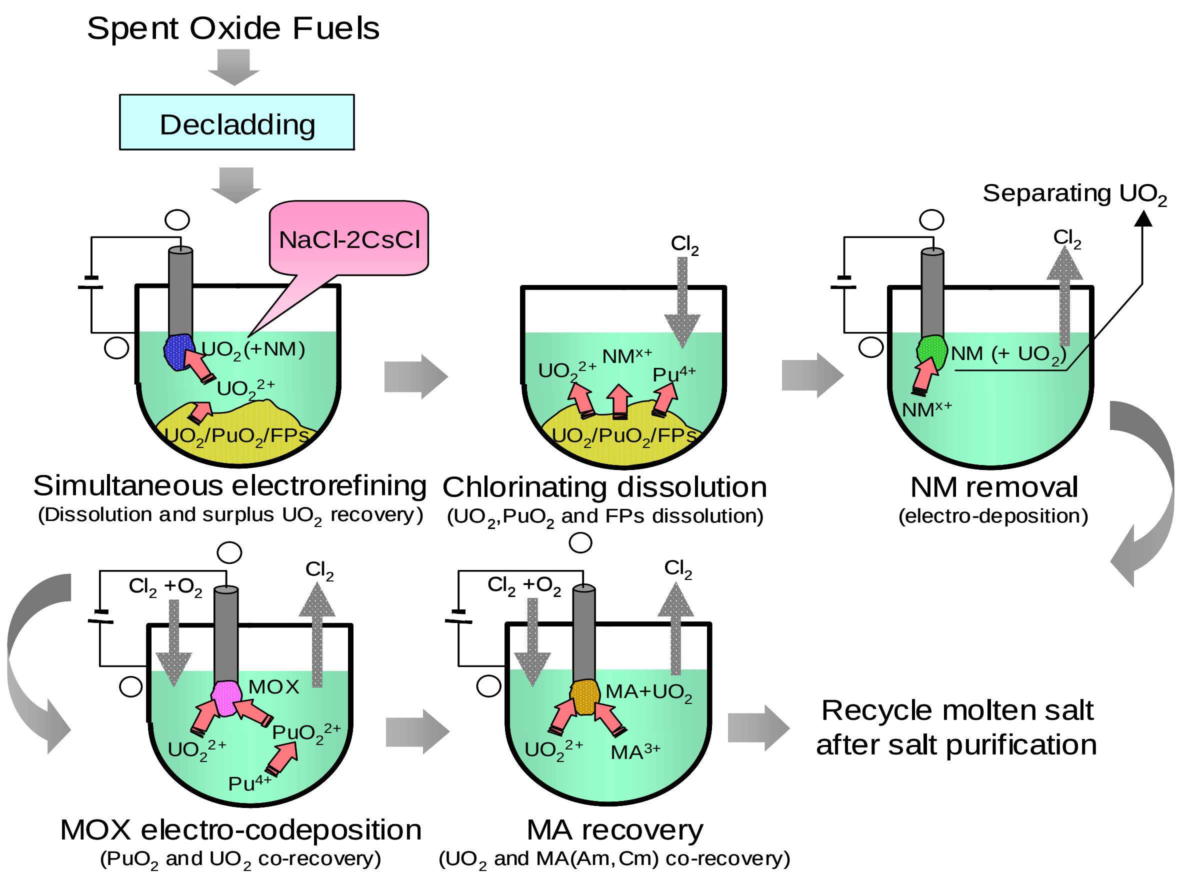

Figure 3. Schematic flow diagram of oxide electrowinning process.

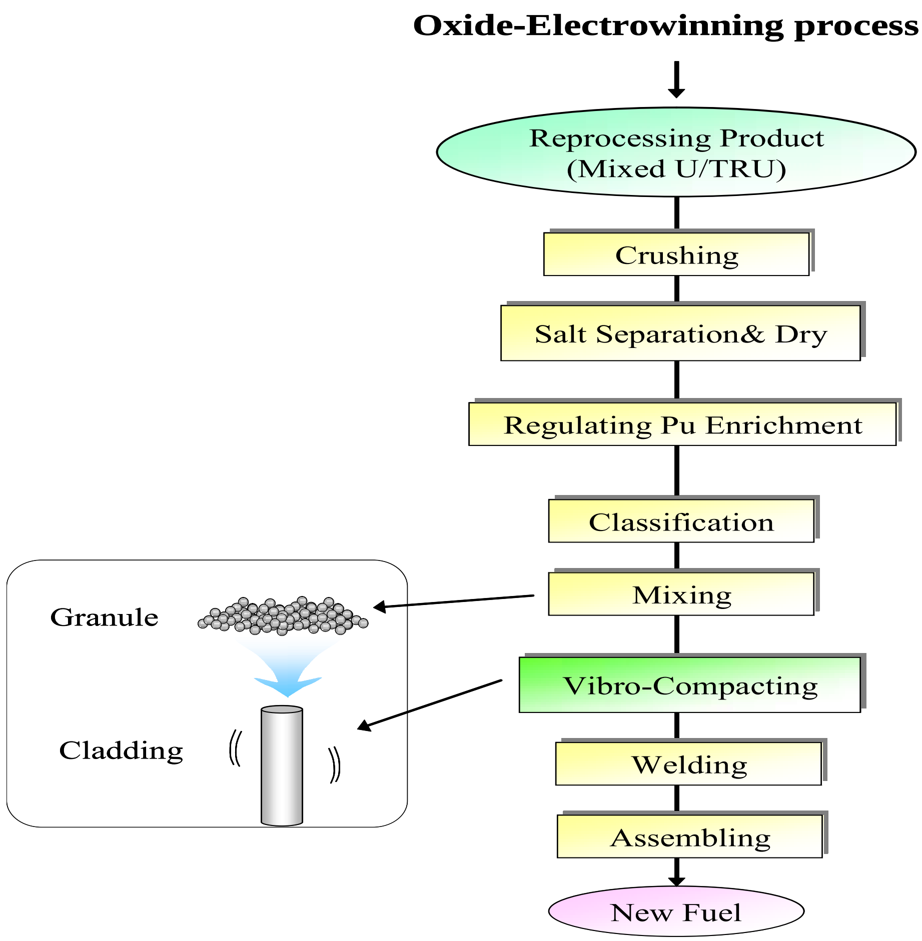

Figure 4. Schematic flow diagram of vibropac fuel fabrication process.

In Japan, a facility sized at 50 MTHM/yr (the total of the core fuel and blanket fuel processing volume) is the basis for evaluation. This corresponds to a yield of spent fuel from about 2.6 reactors (at 1.5 GWe per reactor with breeding ratio of 1.2).

R&D items related to each process step are as follows:

3.1.3.1 Head End Process and Fuel Preparation.

a. Disassembling and shearing

See 3.1.1.1 a.

b. Decladding

Powdering of spent fuel is a valid process for oxide electrowinning. To get powdered spent fuel, a mechanical decladding system such as a shredder has been examined. Powdering performance of spent fuel and separation efficiency of powdered claddings from spent fuel must be studied for this decladding system.

3.1.3.2 *Oxide Electrowinning Process Option.*

a. Anode dissolution and UO_2 deposition

The simultaneous electrowinning (anode dissolution and a surplus-UO_2 deposition process) is effective technology in order to increase the processing speed and reduce the required amount of chlorine. Small-scale uranium tests were implemented. Small-scale hot tests are required for understanding of anode dissolution properties of U and Pu.

b. Chlorinating dissolution

Small scale hot tests are required for understanding the chlorinating dissolution performance of U and Pu, and the volatility of other elements (fission products).

c. Noble metals (NMs) removal

A small-scale uranium test with NMs and a platinum cathode shows that NMs are selectively collected onto the cathode with little uranium deposition. Furthermore, small-scale hot tests are required for understanding the removal performance of NMs.

d. MOX co-deposition

MOX co-deposition behavior has been demonstrated extensively in engineering-scale tests with highly-decontaminated UO_2 and PuO_2. However, it is known that some of fission products could disturb the MOX co-deposition behavior. Therefore, small-scale hot tests are required for understanding U and Pu co-deposition performance.

e. MA recovery

Electrowinning of MAs with U and Pu is a candidate for MA recovery , but only basic data on MA behavior in molten salts have been investigated. Thus demonstration through small-scale hot tests is required for development of MA recovery technology.

f. Electrowinning equipment

Element and engineering scale tests are required for understanding of salt heating/cooling performance and volatile salt behavior. Evaluation of handling under high temperature, corrosion resistance of materials, and materials development for life extension of components (e.g., crucibles) may be required. The life extension of components is effective for cost reduction.

g. Chlorine recycle system

Understanding chlorine recovery performance and development of a chlorine recycle system may be required.

h. Cathode treatment system

Understanding recovery performance and development of a cathode treatment system may be required.

i. Salt separation system

Evaluation of vacuum distillation performance and development of distillation components may be required.

3.1.3.3 Vibropac Fuel Fabrication Option.

Development of vibration-compaction fuel fabrication, combined with the electrowinning method, started in the 1970s at Research Institute of Atomic Reactor in Russia. Up to now, about 18,000 fuel pins have been prepared in a pilot fabrication plant constructed in 1977. The R&D items required are to attain high density, and to develop a non-destructive inspection method for low decontamination fuel. In spite of the Russian irradiation database for this fuel, it will likely be necessary to do steady state and transient testing on vibropac fuel elements.

a. Fuel meat preparation (Classifying/Mixing process/Meat inspection)

As a viability R&D task, the deposit from electrowinning, which is obtained electrolytically in the processing step, will be crushed and classified into a suitable distribution of particle diameters for the vibropac process. Classified granules will be mixed to adjust to an optimal weight ratio. An apparatus based on a small-scale vibro-sieve will be examined. With a U/Pu/MA composition, the granule produced from electrolytic deposition is heterogeneous. Therefore, a granule sampling method must be devised as a viability R&D task, which will include evaluation of the batch representation, sampling quantity, and the inspection technique for quality control.

Performance R&D involves provision of examination equipment for in-cell testing which will be automatically controlled and remotely maintained.

b. Vibropac process

Performance R&D will be carried out on a small-scale vibropac experiment to accumulate vibropac process data such as vibration frequency and acceleration. The main parameters to get a satisfactory fuel specification (involving for example, filling density and axial density distribution), will be evaluated.

Also, an automated vibropac apparatus, intended for remote operation and maintenance, will be designed and manufactured. The performance of the apparatus with respect to such items as processing speed and operational reliability will be examined.

c. ODS cladding welding/welding inspection

See 3.1.1.3 e.

d. Quality control for fuel pins

The inspection technique for the axial density /Pu distribution is to be tested by non-destructive means. The fuel rod tests have to be performed with required accuracy in a short inspection time to confirm the inspection capability in a high-radiation environment..

e. Automated and Remote Material Handling

Performance R&D is needed to assure that process equipment and material handling devices installed in a hot-cell environment can be operated and maintained remotely.

3.1.3.4 Waste Management.

a. High level waste

With regard to phosphate precipitation, which is the process to separate fission products as phosphates from molten salts in addition to phosphoric acid, cold precipitation tests are required for understanding the decontamination efficiency for each element. The vitrification process for phosphate wastes (including fission products) from phosphate precipitation has been studied. In Russia there is significant experience with phosphate glass. An adaptability evaluation for disposal is required.

With regard to waste salts, the vitrification method with borosilicate glass or phosphate glass after oxidation has been studied. Small-scale tests are required for understanding the oxidation efficiency and technical feasibility. An adaptability evaluation for disposal is also required.

b. TRU waste

The reduction and stablization of discarded process equipment including electrolysis crucibles, cathodes etc. from the oxide electrowinning processing has been studied. Improved decontamination technology reduces the waste category to lower one, enabling reduced disposal cost.

3.1.3.5 Process Control and Accountability. The main required R&D is verification of the analysis method and analysis system, and study of safeguard concepts for the advanced non-aqueous process.

The development of an in-situ analysis method including sampling is required as is analysis technology for molten salt. Advancements in the system may be required for large-scale application of this technology. From the aspect of system design, development of a material balance analysis code may be required.

3.1.3.6 Fuel Cycle Safety. The main example of the required R&D is the study of a nuclear criticality safety management method for advanced non-aqueous processes.

3.1.4 Nuclear Fuel Cycle Security

It seems certain that future nuclear fuel recycle technologies will be affected by heightened physical security. From an R&D planning perspective, the issue is not enhanced physical protection itself although that is almost certain to be an element of future nuclear deployments. The R&D issue is to develop technologies and systems (and fuel cycle plant designs) that are inherently easier to secure. The R&D element is little more than a placeholder at the present time.

3.2 Safety

The following discussion defines areas in which technology gaps may exist and in which research and development activities may be necessary to close those gaps.

This plan deals with the technology gaps and required R&D in three categories. The first is basic data. R&D in this area is aimed at providing the basic nuclear, heat transfer, thermal-hydraulic, materials property, and chemical data needed to model and analyze the safety of L1 or L2 systems. The second category is passive safety. R&D in this category is aimed at providing assurance that the physical phenomena and related design features relied upon to achieve passive response to design basis events and those beyond-design-basis events historically analyzed in licensing (i.e., ATWS) are adequately characterized. This R&D will investigate phenomena such as axial fuel expansion and radial core expansion, and design features such as self-activated shutdown systems and passive decay heat removal systems. The third category is bounding events. This R&D will investigate the fundamental phenomena relied upon to mitigate bounding events that may be postulated as ultimate tests of the containment capability of an L1/L2 system. Such phenomena might include draining of molten fuel from a subassembly to preclude recriticality, and coolability of a fuel debris bed under a sodium pool. The R&D is aimed at providing confidence that passive safety extends to bounding events and that a prototype plant poses no unacceptable risk as part of the licensing process.

When dealing with reactor and fuel cycle system concept safety, it is convenient to think in terms of performance of the basic functions necessary to fulfill the fundamental purpose of nuclear safety, that being to protect the public and the plant workers from harmful exposure to radiation. For a reactor, the three basic safety functions are: (1) maintain heat generation and heat removal in proper balance; (2) achieve and maintain reactor shutdown and remove decay heat after reactor shutdown; and (3) maintain appropriate shielding and containment of radioactive materials.

3.2.1 Overpower (Reactivity Insertion) Accidents

A reactivity-driven overpower transient is one principal challenge to maintaining heat generation and heat removal in proper balance. Passive mechanisms to limit such power transients are needed. The well-established Doppler effect in U-238 is one such mechanism. Another is axial expansion of fuel within the cladding. In addition, it is necessary to establish the conditions that will lead to significant fuel failures in overpower accidents so that the adequacy of the passive reactivity feedbacks to terminate a transient without fuel failure can be proven. For risk assessment and licensing purposes, it may also be necessary to investigate the fuel and fission product release associated with fuel failures, as well as coolability of the core after failure.

Basic Data

Technology Gaps: The neutronics data needed for analysis of a reactivity-insertion accident are well established for U-Pu fuels. Some additional measurements may be needed for fuels containing recycled minor actinides, but such data are probably not needed to establish concept viability. Thermal-hydraulic data are well known for sodium coolant. No additional R&D is needed for basic data.

R&D Tasks: None

Passive Safety

- Pre-failure Fuel Relocation

  Technology Gaps: Axial expansion of fuel prior to failure will remove reactivity and turn a reactivity-insertion-driven power transient. The magnitude of the effect is quite different for oxide and metal fuels. Prefailure fuel relocation has been extensively investigated for oxide fuels over the years. No additional R&D is recommended. For metal fuels, axial fuel expansion is one of the primary passive mechanisms, along with radial core expansion and Doppler feedback, relied upon to remove reactivity, restore the reactor to criticality, and limit power increase. The magnitude and dynamics of axial fuel expansion over a range of conditions expected for design basis and beyond design basis postulated initiating events (PIEs) must be investigated to support modeling and code validation. Some data are available from IFR TREAT experiments. Additional experiments are needed to extend the range of data and investigate margins to failure.

  R&D Tasks:

  - S1.1a: In-pile experiments simulating overpower transients using the Transient Reactor Test (TREAT) facility are needed to extend the range of the existing data for metal fuels, and to investigate margins to failure. These experiments should use fuel that is as prototype as possible; that is, ternary U-Pu-Zr fuel and ferritic steel (D9) cladding, with a range of burnups. A minimum of three experiments is considered necessary to investigate pre-failure fuel relocation, using fuel of low, intermediate, and high burnup. The experiments should simulate ATWS events, stopping short of predicted fuel pin failure.

- S1.1b: Data from the pre-failure fuel relocation tests should be used to validate code predictions of fuel behavior during overpower transients.

Bounding Events

- Fuel Failure and Fission Product Release

Technology Gaps: Fuel failure and fission product release has been extensively investigated in many experiments for oxide fuels. This extensive database has been used to develop and validate analytical models. No additional R&D is recommended. While it is expected that passive response in a metal-fueled system can preclude fuel failures in all but the most unlikely PIEs, it is necessary to extend the range of existing overpower experiments to determine the failure power level and energy, to investigate the extent of fuel release and relocation after failure, and the extent of fission product release after failure. Only limited data are available for metal fuel.

R&D Tasks:

- S1.2a: Fuel failure and fission product release in overpower accidents should be investigated in TREAT experiments similar to those for pre-failure fuel relocation. The principal difference is that the power transient should be extended to obtain fuel pre-failure, with subsequent fuel relocation out of the failed pins, along with fission product release into the coolant. A minimum of three experiments is considered necessary, using low, medium and high burnup fuel of prototype design, similar to the fuel pins tested in the pre-failure fuel relocation series.

- S1.2b: Data on fuel pre-failure power and energy, along with observations of fuel relocation following failure and fission product release should be used to validate code predictions of these phenomena. In addition, the fission product release data should be used to establish a source term for evaluation of fission product transport in the coolant and release to the containment.

- Post-accident Coolability

Technology Gaps: Existing experiments on oxide fuel are adequate to evaluate coolability of debris. For metal fuels, data are needed to verify that damaged fuel and fuel debris are coolable in liquid sodium. Only very limited data are available for metal fuels.

R&D Tasks:

- S1.3a: The experiments from task S1.2a, above, should be examined to determine the coolability of the damaged fuel elements. However, because of the hydraulic characteristics of the TREAT apparatus, the results may not be definitive.

- S1.3b: Since TREAT experiments will have non-prototype hydraulic characteristics, it will likely be necessary to conduct additional out-of-pile experiments in a loop having more prototypic hydraulic conditions. The Components and Materials Evaluation Loop (CAMEL) facility has been used for this purpose, and is recommended for this task. Injection of molten fuel into a pin-bundle with flowing sodium is used to simulate fuel pin failure with ejection of fuel from the cladding. Transport of the molten fuel and evaluation of the remaining flow after failure are the key observations.

3.2.2 Undercooling Accidents

Loss-of-coolant-flow and loss-of-heat-sink transients are the other major class of challenges to the proper heat generation/heat removal balance. Passive response of the reactor core and heat transport systems to restore the balance is needed. The overall viability of such response in a pool type, sodium-cooled, metal-fueled system has been demonstrated in the landmark EBR-II experiments of April, 1986. Additional R&D work is aimed at generalizing the understanding of the feedback mechanisms and investigating alternative shutdown concepts, as well as supporting risk assessment and licensing. For the oxide-fueled L1 systems, the reactivity feedbacks will be somewhat different because of the larger core and the larger Doppler "return" effect as power decreases. More reliance on self-actuated shutdown systems or gas-expansion modules (GEMs) may be necessary.

<u>Basic Data</u>

<u>Technology Gaps</u>: Generally, the basic data needed to analyze undercooling accidents are adequate, with possible exception of data for fuel containing minor actinides. No additional R&D is recommended. See Section 3.2.1.

<u>R&D Tasks</u>: None.

<u>Passive Safety</u>

- Fuel and Core Expansion

<u>Technology Gaps</u>: A dominant feedback mechanism in an undercooling transient without scram will likely be axial and radial core expansion and bowing of subassemblies. Only very limited data on these phenomena are available, and analytical capabilities are also limited. Development of coupled thermal-hydraulic-structural analysis tools is needed, and innovative experiments may be needed to provide data for validation. However, the importance of various reactivity feedback mechanisms will depend on details of the reactor and system design, so it is not possible to identify definitive experiment needs at this time. Because of the complexity of the situation, experiments in a prototype reactor may be necessary to finally validate code predictions. This technology gap applies to both oxide and metal-fueled systems.

<u>R&D Tasks</u>:

- S2.1a: An evaluation of the need for experiments to validate predictions of various reactivity feedback mechanisms, such as axial and radial core expansion, subassembly bowing, and control rod driveline expansion should be conducted. In addition, the feasibility of performing such experiments in an out-of-pile facility should be evaluated. With the results of this evaluation, future R&D an be planned.

- S2.1b: Design and perform experiments judged to be needed and feasible in the evaluation conducted under task S2.1a.

<u>Bounding Events</u>

- Fuel Failure and Fission Product Release

Technology Gaps: It is necessary to investigate the response of fuel in transients resulting from very remote PIEs to determine the likelihood of fuel failure and the extent of any fuel release, relocation within and out of a damaged assembly, and fission product release. This area has been adequately investigated for oxide fuels. However, no data are available for metal fuel, and suitable analysis tools are needed.

R&D Tasks

- S2.2a: In-pile experiments simulating undercooling transients using the TREAT facility are needed to establish a data base for predictions of pre-failure fuel behavior, fuel failure, and fission product release. These experiments should use fuel that is as prototype as possible, with a range of burnups. A minimum of three experiments is considered necessary to investigate pre-failure fuel behavior, using fuel of low, intermediate, and high burnup. These experiments should simulate ATWS events terminated short of predicted fuel pin failure. An additional three experiments are recommended, in which the experiment is continued until fuel failure occurs, and post-failure fuel behavior and fission product release can be observed.

- S2.2b: Data from the TREAT experiments in task S2.2a should be utilized to validate code predictions of pre- and post-failure fuel behavior and fission product release during undercooling accidents.

- Post-accident Coolability

Technology Gaps: This area has been adequately investigated for oxide fuels. However, only limited relevant data are available for metal fuels. Suitable experiments are needed to establish coolability of a damaged assembly and debris resulting from a severe undercooling accident.

R&D Tasks:

- S2.3a: The results of the TREAT experiments conducted under task S2.2a should be evaluated for evidence to establish coolability of subassemblies damaged in an undercooling accident.

- S2.3b: If sufficient evidence of subassembly coolability is not available from the TREAT experiments of task S2.2a, it may be necessary to design and conduct additional experiments in the out-of-pile CAMEL facility. This facility will provide a more prototypic hydraulic simulation, and can be used to investigate the effects of coolant reentry on coolability.

3.2.3 Decay Heat Removal

Removal of decay heat in the long term is the second of the basic safety functions. Passive means to transport heat from the core and near environs to an ultimate heat sink is needed. Concepts typically involve natural convection flow through the reactor core, with heat removal from the primary vessel or heat transport system using heat transport through the system boundaries to an air heat sink (i.e., the RVACS concept) or using a passive heat transport loop (i.e., the EBR-II shutdown coolers). More than one heat transport path may be provided to assure redundancy, diversity, and to accommodate decreasing heat load with time. The issues in decay heat removal do not depend on fuel type to first order.

Basic Data

Technology Gaps: Basic data required to evaluate decay heat removal are generally adequately known. No further R&D is recommended.

R&D Tasks: None.

Passive Safety

- Heat Transport in Vessel and Primary HTS

  Technology Gaps: The fundamental mechanisms of heat transport out of the core region to the heat sink are well known. Development of suitable analytical tools may be needed for some systems.

  R&D Tasks: None

- Heat Transfer to Air Heat Sink

  Technology Gaps: Mechanisms for heat transfer through the primary system boundary to an air heat sink are well known in general. Specific data such as emissivity measurements for specific materials or natural convection heat transfer coefficients in particular geometries may be needed for some designs, but given such data, RVACS-type systems can be designed with high confidence. Generally, the performance of actual systems will be evaluated at the startup testing phase for a prototype. No fundamental R&D is recommended, although specific measurements such as mentioned above may be necessary for specific designs.

- Heat Transfer in Passive Heat Transport Loops

  Technology Gaps: The technology of a liquid-liquid heat exchanger, as intermediate loop, and a liquid-to-air heat exchanger is well known. No R&D is recommended.

  R&D Tasks: None.

Bounding Events

- Postaccident Coolability

  Technology Gaps: In general, bounding events will produce core debris which must be cooled for the long term. Post accident coolability has been discussed for overpower and undercooling accidents in 3.2.1 and 3.2.2, respectively. If debris can relocate after an accident, it will be necessary to establish that the debris is coolable in its final location, such as in a debris bed in the inlet plenum. Questions of debris bed coolability have been studied for both oxide and metal fuels, with much more extensive data available for oxide. Coolability of metal-fuel debris beds requires demonstration.

R&D Tasks:

S3.1a: Long-term coolability of core debris resulting from fuel failures in a bounding event is given a top priority in the R&D tasks. Out-of-pile experiments in which molten fuel is introduced into a pool of sodium and the resulting debris bed is characterized are considered necessary. Data from such experiments can be used to establish coolability of the debris bed. A coolable condition is necessary to show accident termination, without continuing damage to additional fuel or structures.

S3.1b: Data from TREAT experiments conducted in tasks S1.2b and S2.2a should be evaluated for insights into relocation of fuel debris and the properties of the debris. This evaluation should emphasize long-term coolability.

3.2.4 Reactor Shutdown

By adopting active shutdown systems with sufficient redundancy and diversity, the frequency of accident sequences leading to core damage can be reduced to a negligibly low level. In addition, even higher shutdown reliability can be provided if desired or if necessary in the individual national licensing arenas by introducing passive safety systems, such as a self-actuated shutdown system (SASS), or a gas expansion module (GEM).

Basic Data

Technology Gaps: Basic data required to evaluate reactor shutdown are generally adequately known. No R&D is recommended.

R&D Tasks: None.

Passive Safety

- Self-actuated Shutdown System (SASS)

Technology Gaps: By adopting a Curie point magnet in a magnetic circuit, absorber rods are de-latched in all types of anticipated-transient-without-scram (ATWS) scenarios. These scenarios include the classical "unprotected" (i.e., failure to scram automatically) accidents: unprotected loss of flow (ULOF); unprotected transient overpower (UTOP), and unprotected loss of heat sink (ULOHS). These transients result in an increase in core outlet temperature, which then can bring the Curie point latches into play if the magnet temperature exceeds a certain value. The SASS concept can be applied to any reactor core concept, not only MOX fueled but also metallic fueled cores. Comprehensive development of the temperature sensitive alloy and demonstration of the effectiveness of SASS are required.

R&D Tasks:

- S4.1a: Tests of the temperature sensitive alloy to evaluate thermal aging, thermal fatigue and creep, thermal transients and effects of irradiation are recommended.
- S4.1b: Transient tests using full-scale prototype SASS devices in a suitable sodium loop in order to confirm the effectiveness of the SASS are recommended.
- S4.1c: A full scale in-pile test of a SASS system to establish its long-term performance in an irradiation environment is recommended.

Gas Expansion Modules (GEMS)

Technology Gaps: GEMS provide an alternative diverse, passive shutdown mechanism. A gas filled chamber is placed in a region of the core having a negative void worth. Upon increase in coolant temperature or decrease in inlet pressure, the gas expands, removing reactivity from the system. The basic technology of the GEM is well known. No R&D tasks are recommended.

Bounding Events

Technology Gaps: While SASS and GEMS may have an important role in bounding event sequences, there are no specific technology gaps associated with bounding events. Therefore, no R&D for bounding events related to reactor shutdown is recommended.

3.2.5 Maintaining Cladding Integrity

So long as the first two basic safety functions are maintained there should be no challenge to the third safety function, maintaining containment of fuel and fission products, provided there is no propagation of random fuel failures at normal conditions. The cladding is the first physical barrier to release of fuel and fission products. Cladding failure therefore degrades defense-in-depth. While a few random cladding failures may be unavoidable, it is important that localized failures not propagate into larger regions of cladding failure and release of appreciable radioactive material. Cladding failure may propagate through thermal or chemical mechanisms.

Basic Data

Technology Gaps: Basic data necessary to evaluate cladding integrity and failure propagation issues are generally well known. The presence of minor actinides in the fuel may introduce some fuel performance issues, but these are dealt with under fuel development. No additional R&D for safety is recommended.

R&D Tasks: None

Passive Safety

- Cladding Failure and Failure Propagation

Technology Gaps: It is necessary to establish that individual fuel element failures will not cause failures of adjacent fuel elements due to disruption of heat transfer and overheating of the adjacent fuel. There has been no evidence of pin-to-pin failure propagation during operation of metal-fueled systems. Modern analytical techniques, such as computational fluid-dynamics (CFD) may be used to evaluate the question of short term disruption of heat transfer. For the oxide-fueled-system, there is experimental evidence from the EBR-II run-beyond-cladding breach program that pin-to-pin failure propagation may occur under aggressive operating conditions if pins are run well beyond the time of detection of fuel release. Methods for detection of fuel failures are well known. No additional R&D is recommended.

- Fuel-cladding Chemical Interaction

Technology Gaps: It is necessary to establish that fuel element failure propagation will not occur by chemical mechanisms. For oxide fuels, it is known that interaction between the fuel and coolant will cause swelling of the fuel and a potential failure propagation mechanism. More accurate prediction of cladding failure is needed to advance the fuel design. For metal fuels, the fuel-coolant chemistry is generally well known and is benign. Addition of minor actinides to the fuel mixture will introduce an uncertainty. More data on high burnup fuel/clad chemical interaction of recycled fuel is needed. This is a fuel development issue. No substantial safety-related R&D is recommended.

R&D Tasks: None.

Fission Product and Coolant Chemistry

Technology Gaps: It is necessary to establish that chemical interaction between the coolant and fission products released upon cladding failure is such that release can be detected, clean up can be performed, and that the potential for fission product release to the containment is minimized. The chemical affinity of sodium coolant for fission products of high interest in this connection (i.e., iodine, cesium, strontium) is well known. This chemical affinity provides an important mechanism for mitigating fission product release. Similarly, the technology of detection of fuel failure and cleanup of sodium coolant systems is well known, and no associated R&D is recommended.

R&D Tasks: None.

3.2.6 Maintaining Containment Integrity

For the systems under consideration (L1/L2), it should not be possible to find a mechanistic initiating event of any appreciable probability that would lead to a severe accident involving core melting and a challenge to the containment. Extremely low probability combinations of events and non-mechanistically postulated situations may be used to test the containment design for licensing purposes and to assess residual risk. Generically, threats to containment could come from rapid internal energy release such as might be associated with an energetic recriticality, or from long-term pressurization of the containment such as might result from thermal or chemical interactions between fuel, sodium, and concrete producing non-condensable gases. Thus, it is necessary to establish the design requirements for the containment including consideration of both long-term static pressure capability and the ability to withstand short-term dynamic loadings.

Mechanisms for long-term containment pressurization include chemical interactions between core materials, sodium, and concrete or other containment materials. Areas for related R&D include: (1) evaluation of in-vessel coolability of core debris; (2) chemical interactions between core debris, sodium, and concrete; and (3) the consequences of sodium leaks and fires.

The principal concern in short-term loading is the possibility of an energetic recriticality due to compaction of material in a damaged core. Historically, this issue has received a great deal of attention in licensing proceedings for fast reactors. Much related analysis and experimentation has been directed at this issue for oxide fueled systems. As a result of this work, the likelihood of an energetic recriticality has been deemed to be very low, but designers and regulators have not been able to definitively exclude the possibility. Thus, in Japan, a "recriticality free" core concept has been introduced as a special design feature, intended to further reduce the likelihood of energetic recriticality. R&D associated with this concept is included under "Special Topics" below.

For a metal-fueled system, the same fundamental potential for recriticality exists. However, because of the low melting point of the fuel, it is expected that enough molten fuel can be removed from a damaged core by draining downwards to render the core permanently subcritical and preclude recriticality. The potential for recriticality in a debris bed appears to be low for metal fuel due to spreading of the melt and low density of the solidified debris bed. Both assertions remain to be proven experimentally.

Basic Data

Technology Gaps: The basic data needed to evaluate the issues of containment integrity are generally well known. No related R&D is foreseen.

R&D Tasks: None

Passive Safety

Technology Gaps: If the passive safety features of the reactor and system design are successful, there will be no issues of containment integrity for the L1/L2 systems. While sodium leaks and fires may properly be considered to be within the design basis, the technology for prevention, detection, and control of such events is well known. Similarly, the technology of dealing with sodium-water reactions upon steam generator tube leaks is well known. These are design issues. No R&D in containment integrity is recommended.

R&D Tasks: None.

Bounding Events

- In-vessel Debris Retention

  Technology Gaps: In-vessel retention of core debris is an extremely important issue because in some designs, breach of the reactor vessel or a guard vessel is a containment failure. In addition, if in-vessel retention can be shown, issues of sodium-concrete reactions, core debris-concrete reactions, and the like are rendered moot.

  Coolability of debris in an oxide-fueled sodium system has been investigated experimentally. These results should be reviewed to determine if they support in-vessel debris coolability. If not, some limited work in this area may be necessary.

  Some limited work was done on coolability of metal fuel debris under sodium with promising results. These data should be reviewed for applicability. Limited viability R&D is considered necessary to confirm in-vessel debris retention.

  R&D Tasks:

  - S6.1a: The issue of in-vessel debris retention is closely related to the issues of debris coolability addressed in connection with gap S3.1. However, the implications of in-vessel retention with respect to containment integrity make it important to focus separately on this issue. However, the experiments conducted under task S3.1a may provide useful and pertinent data, so they should be examined as a top priority task.

  - S6.1b: Out-of-pile experiments involving mixing molten fuel and sodium coolant under prototypic conditions of geometry, temperature, and quantities will likely be necessary to resolve issues of in-vessel coolability. These experiments may be similar to those of task S3.1a, but should be performed and analyzed with particular attention to in-vessel retention issues.

- Sodium-concrete Interaction

  Technology Gaps: This area has been adequately investigated and no additional R&D is recommended.

  R&D Tasks: None.

- Sodium Leaks and Fires

  Technology Gaps: The technology related to sodium leaks and fires is well known. This area is considered to be a design issue, and no additional R&D is recommended.

  R&D Tasks: None.

- Post-failure Fuel Relocation

  Technology Gaps: This is the key issue in assessment of recriticality. It has been extensively investigated in many experiments for oxide fuels. As noted above, the results are not definitive, so the JNC has introduced the "recriticality free" concept for L1. R&D to prove the viability of this concept is included under "Other Special Topics." Only limited analytical investigation has been performed for metal fuels. Therefore, additional modeling and analysis, supported by at least limited laboratory and/or in-pile experimental work is recommended. Since recriticality has been an important issue in licensing of LMRs, a definitive technical position is considered necessary. Resolution of this issue is considered to be viability R&D.

  R&D Tasks:

  - S6.2a: Analysis, modeling, and code development is required to address the issues of fuel removal from a damaged core to exclude the possibility of recriticality. While the most likely path for fuel transport is downwards into the lower vessel plenum, removal upwards into the outlet plenum should also be considered.

  - S6.2b: Suitable out-of-pile experiments should be designed and performed to guide the modeling effort of task S6.2a, and to provide data for code validation. These experiments will likely involve prototypic geometrical mock-ups of inlet hardware, sodium coolant, and molten fuel at prototypic temperature and composition.

  - S6.2c: Another issue in recriticality analysis is spreading of a debris bed in the lower plenum to avoid the possibility of recriticality in the debris. Data on melt spreading and debris bed formation from task S3.1a and other sources should be reviewed for adequacy.

  - S6.2d: Additional out-of-pile experiments designed to address melt spreading and debris bed formation, with particular attention to showing that recriticality can be excluded, may be needed. Such experiments will be closely related to the fuel removal experiments of task S6.2b, and may involve addition of a suitable geometric mock-up to those experiments.

3.2.7 Safety Analysis Methods and Codes

Although there are many well-developed safety analysis methods and codes available and applicable to the L1 and L2 systems, it is necessary to upgrade their modeling and make use of modern computing capabilities. In addition, modern approaches, such as CFD, may allow assessment of some questions by analysis where experiments were previously needed. An example might be assessment of the heat transfer perturbation associated with a fuel pin failure. New probabilistic techniques may be necessary to properly account for passive safety features of the L1/L2 designs.

Basic Data: Not applicable.

Passive Safety

- Deterministic Methods and Codes

  Technology Gaps: Most methods and codes applicable to L1/L2 safety analysis were developed for oxide-fueled systems. Some of these codes have been extended for metal fuels, with a limited range of applicability and validation. With respect to passive safety, it is necessary to extend existing modeling of the thermal-hydraulic-structural behavior of a core to improve the ability to predict radial core expansion and assembly bowing in loss of flow situations. Radial expansion and bowing is a major contributor to the passively safe response of a L2 system to a loss of flow, and high confidence in its efficacy must be developed.

It is expected that the Argonne-originated SAS codes will be the major focus of code development. The next step in development for this purpose is multiple-pin modeling to obtain an estimate of the temperature gradient across the subassembly which determines the duct temperature and its mechanical response. This calculation requires pin-to-pin power distributions from the physics calculations and channel-by-channel thermal hydraulics, integrated with a subassembly-by-subassembly mechanical calculation. Reduction in uncertainties in reactor physics data will also be necessary.

Data for validation of the next-generation SAS code is available from the EBR-II SHRT tests, the FFTF inherent safety tests, and various TREAT tests. Data from core restraint tests done at Westinghouse for the CRBRP project may also be useful. However, additional data will almost certainly be necessary (see Section 3.2.2).

R&D Tasks:

- S7.1a: Analysis, modeling, and code development based on the existing SAS code should be undertaken to improve the capabilities in that code to model the phenomena most important in showing passive safety for DBAs and ATWS events. Areas of emphasis are discussed above.

- S7.1b: Code validation activities should be pursued in parallel with development work. Data from existing sources, plus that from R&D associated with gaps S1.1, S1.2, S2.1, and S2.2 should be used as available.

- Probabilistic Methods and Codes

Technology Gaps: The Generation IV systems, and L1/L2 in particular have a very high emphasis on passive safety. Probabilistic methods and codes are needed to analyze such systems to assure that the probabilities of failure of passive safety features leading to core damage is indeed acceptably low. Existing methods and codes are probably adequate for the stage of development considered in this document, so no near-term R&D is recommended.

R&D Tasks: None

Bounding Events

- Deterministic Methods and Codes

Technology Gaps: Methods and codes for bounding events have been developed extensively for oxide-fueled systems over the years. This work is applicable to the L1 systems. However, much less work on metal-fueled systems has been done. While the need for extensive modeling and code development is diminished by the passive safety attributes of the L2 systems, it is nonetheless necessary to develop models to analyze key issues in bounding events. In particular, it is necessary to develop models to predict fuel relocation out of damaged subassemblies so that the potential for re-criticality can be assessed.

R&D Tasks:

- S7.2a: The current modeling in the SAS codes does not include all of the models needed for analysis of bounding events. Such models are needed to complete the safety analysis capability. Adequate models of fuel failure and fuel relocation should be a principal goal.

- S7.2b: Validation of models developed under task S7.2a should be conducted using data from R&D associated with gaps S1.2, S1.3, S2.1, S2.2, S2.3, S3.1 and S6.2, as appropriate.

3.2.8 Other Special Topics

By providing passive safety features to guard against rapid event sequences such as ATWS events, and redundant accident prevention features to guard against slow sequences, the probability of core disruptive accidents (CDAs) can be made negligibly small. Nevertheless, in some national fast reactor programs, mitigation of potential CDA consequences is seen as potentially desirable. In particular, potential for re-criticality and a resulting power excursion in the course of CDA is regarded in Japan as one of the major safety issues in fast reactor development.

The potential for energetic re-criticality in the initiating phase (I/P) of a CDA sequence can be eliminated by limiting positive void worth. Enhancing molten fuel discharge from the core region (to prevent large-scale molten pool formation) will eliminate re-criticality during the transition phase (T/P). Long-term coolability of the relocated fuel debris inside the reactor vessel must be secured, possibly by providing several design measures.

In the course of ULOF accident, power excursions due to coherent sodium boiling have been of concern, because a large sodium-cooled core tends to have a large sodium void worth. In order to avoid any possibility of severe energetics during the I/P, the sodium void worth should be limited. Based on theoretical considerations and experimental data on oxide-fueled systems in Japan the reference value of the maximum sodium void worth can be set 6$ for a MOX fueled core and 8$ for a metallic fueled system.

As for the fuel discharge in the T/P, special fuel subassemblies have been proposed in Japan to enhance discharge of the molten fuel. The fuel assembly with an inner duct structure (FAIDUS) has been selected, because the fuel discharge process would be nearly one-dimensional. It is thought to have less uncertainty than alternatives. The experimental program named EAGLE was launched in 1998 as collaboration between Japan and the Republic of Kazakhstan. Both in-pile and out-of-pile experiments are now in progress. The FAIDUS assembly also requires several R&D items related to fabrication and irradiation integrity.

Another concept (ABLE, for Axial Blanket partially Eliminated) is expected to improve the core performance over that of FAIDUS, and less R&D is required for design of the assembly. However, the fuel discharge process of ABLE contains more uncertainty and needs much time to complete its study. Analytical efforts are now underway. After a detailed understanding of the fuel discharge process in ABLE is achieved, the experimental requirements will be clarified.

R&D Tasks:

- S8.1a: Conduct suitable in-pile experiments to demonstrate the molten fuel discharge capabilities of the FAIDUS concept.

- S8.1b: Conduct suitable experiments to demonstrate the molten fuels discharge capability of the ABLE concept.

3.3 Fuel Development

The goal of a fuel development program is to develop and demonstrate fuel fabrication technology and fuel designs for irradiation performance that will allow economical, safe, and predictable behavior during fabrication and service. Specific objectives of such a program are:

- A reference fuel design (embodied in a "Fuel Specification")
- A fuel performance database (the results of which are incorporated into a predictive fuel performance code that supports licensing and core design)
- A safety case for operation of a specific reactor, to operate with the reference fuel (which can be incorporated into a licensing case).

R&D activities are initially directed toward developing designs that accomplish general goals with regard to, for example, burnup capability or fabrication losses. After initial designs are determined, then further R&D is used to evaluate limiting conditions of operation and to incorporate design improvements or enhancements. Sufficient performance data and fabrication experience must be obtained to support the establishment of a safety case for the operation of a reactor with a core-load of the new fuel. Eventually, when a reactor begins operation with the new fuel design, lead assemblies are irradiated under surveillance in controlled manner – evaluation of the behavior of these lead assemblies ensures that the fuel behaves in service as was assumed for the safety case, validating continued operation with the safety authorization basis.

In general, fuel development needs and activities can be classified into four categories, as described below.

- Property Determination: The basic thermophysical properties and material characteristics of the fuel and cladding materials must be determined to support initial R&D activities, and with subsequently better precision to reduce uncertainties in safety analysis for licensing.
- Fabrication Development: Economically acceptable techniques for fabricating the fuel must be developed and process repeatability demonstrated. For some fuel systems, complications associated with minor actinide contents must be addressed. These might include achieving desirable stoichiometries for multi-valent ceramic fuels or suppressing americium volatility for fuels with high-temperature fabrication processes. Other issues include remotization of processes for which most experience is with equipment that can be accessed for hands-on maintenance and/or operation.
- Irradiation Performance: The in-service performance of any fuel system must be meet burnup and reliability requirements driven by safety and economic considerations. Therefore, fuel behavior and reliability under nominal and off-normal conditions must be assessed through in-reactor and out-of-pile testing. Life-limiting phenomena, and behavior that otherwise impacts the ability of the fuel to function as required, are identified and their extents established as a function of service conditions. Any unsuitable performance is addressed by further development and design improvements.
- Modeling and Code Development: Licensing of a reactor that will operate with a new fuel design, by established practice, requires the development and implementation of predictive behavior models, typically incorporated into a fuel performance code. Initially, model development is focused on aiding the understanding of life-limiting phenomena. However, once validated, those

models are integrated into other models and codes that will help interpret and eventually predict whole-pin and whole-assembly behavior.

Specific activities identified for continued development of metal and nitride fuels are addressed below.

3.3.1 Metal Fuel (U-Pu-Zr)

In general, the performance of U-Pu-Zr (the base composition of the reference metal fuel form) is sufficiently characterized that viability is assured. However, proposed fuel cycle applications of sodium cooled fast reactors call for recycle of spent fuel, and this gives rise to a requirement for additional viability R&D. The characteristics of recycled fuel must be evaluated to determine whether such fuel operates within the established performance database for the base U-Pu-Zr composition. Other complications with recycled fuel, in fabrication for example, are expected due to the presence of residual fission products and enhanced quantities of minor actinides. Therefore, the viability R&D called for in this plan is characterized by the determination of whether recycled fuel fabrication and performance is sufficiently within previous experience, and for aspects in which it is not, by the improvement to the reference design or fabrication process.

3.3.1.1 Property Determination.

Technology Gaps

The properties of U-Pu-Zr must be better determined to support licensing and to reduce uncertainties in safety analyses of whole-core behavior, which can lead to limits on operation. General R&D tasks required for eventual fuel licensing include the following:

- Evaluation of temperature-dependent thermophysical properties.
- Further evaluation of the interdiffusion behavior of fuel constituents, which leads to zirconium depletion in the fuel microstructure, to determine the conditions and limits.
- Further assessment of phase equilibria in the U-Pu-Zr system.

The impacts of minor actinide additions on properties must be assessed. However, establishment of viability requires only a small number of confirmatory measurements to confirm that key properties of recycled fuel are similar to those of the base U-Pu-Zr.

R&D Tasks

- Confirmation of thermal conductivity, heat capacity, and thermal expansion of recycled fuel.
- Investigation of fuel/cladding interdiffusion behavior with enhanced quantities of rare earth fission products and minor actinides.

3.3.1.2 Fabrication Development.

Technology Gaps

The feasibility of injection casting of U-Pu-Zr has been established and the fabrication of U-10Zr well established at the engineering-scale, production level. However, improvements to the injection casting process must address issues as described below. Furthermore, development of advanced fabrication techniques should be pursued.

- Reduction or elimination of fabrication losses of actinides, through development of new crucible designs and re-usable molds for fuel slugs.

- Suppression or accommodation of volatile americium loss during induction melting, alloying and casting operations for U-Pu-Zr with enhanced minor actinide contents.

  Development of advanced fabrication techniques that do the following:

- Facilitate remote operations

- Eliminate the need for molds and/or crucibles

- Because successful utilization of sodium-cooled reactors will require efficient recycle of actinides within the fuel cycle, fabrication technologies that minimize or eliminate loss of actinides must be developed and demonstrated as viability R&D.

<u>R&D Tasks</u>

- Modification of the reference injection casting fabrication process to suppress americium volatilization or to recover volatilized americium for re-incorporation into the fabrication process; or

- Development of alternative fabrication processes (such as arc casting) that avoid or reduce americium volatilization.

- Development of reusable injection casting molds, or

- Development of alternative fabrication techniques (such as continuous casting) that do not require molds.

3.3.1.3 Irradiation Performance. The irradiation performance of U-Pu-Zr has been established for nominal steady-state conditions and is acceptable. Furthermore, performance under transient overpower and loss-of-flow conditions has been assessed, but further testing under more specific conditions will likely be required prior to preparation of safety case for a specific reactor.

- R&D issues that must be addressed in order to develop a safety case for reactor use of recycled fuel are listed below. However, not all of the identified issues are necessary to establish viability.

- Qualification irradiations of U-Pu-Zr, including 2-sigma conditions, to ensure that U-Pu-Zr behavior is bounded by a safety analysis for whole-core operation.

- Evaluation of the effect of enhanced minor actinide and recycling impurity contents on irradiation performance lifetime.

- Transient testing of high-burnup U-Pu-Zr, with and without minor actinide and reprocessing impurity additions, under specific transient overpower and loss-of-flow conditions.

- Development of cladding alloys (or identification and evaluation of existing alloys) with improved high-temperature strength over HT9, but with similar resistance to swelling.

- Development of cladding liners that will reduce effects of fuel-cladding interdiffusion, or improved cladding alloys that incorporate this feature in addition to high-temperature strength and swelling resistance.

Technology Gaps

The effects of minor actinide additions or recycling impurity carry-over on high-burnup performance have not been fully assessed, nor has behavior at 2-sigma conditions of power and temperature. Demonstration of acceptable performance of recycled fuel is particularly important, because rare earth fission products that accumulate over the course of irradiation in the base U-Pu-Zr fuel have been shown to exacerbate fuel/cladding interdiffusion (FCCI). Furthermore, as is true with most diffusion phenomena, increased temperature enhances the kinetics of FCCI, such that potential fuel-cladding interface temperatures must be assessed for steady-state and off-normal operations to ensure that temperature effect on FCCI remains acceptable. FCCI can lead to an effective localized thinning of the cladding and to formation of low-melting-temperature phases at the fuel-cladding interface. Therefore, the presence of residual rare earth fission products in as-fabricated recycle fuel implies a potential for enhanced FCCI, which would also be temperature dependent. The FCCI phenomenon proved to be life limiting in the base U-Pu-Zr fuel, and provided the basis for time-temperature limits associated with design-basis accidents; however, the necessary limiting conditions of operations still allowed operation of the reactor (EBR-II, in this case) at the desired conditions. The acceptable behavior of recycled fuel with residual rare earth fission products must be investigated to establish viability of this fuel, and design improvements developed, if necessary for viability.

Fuel constituent migration was observed with irradiation of the base U-Pu-Zr, The migration led to the formation of Zr-enhanced and Zr-depleted zones. The purposes of the Zr alloying addition to the U-Pu alloy are to increase the fuel alloy solidus temperature to above that of U-Pu and to mitigate the extent and consequences of FCCI. Therefore, the concern over such zone formation was the potential for transient heating (during an off-normal event) of a Zr-depleted zone, which would have a relatively lower melting temperature, or the potential for low-Zr fuel to contact the cladding with subsequent enhanced FCCI. The presence of appreciable quantities of minor actinides in the fuel alloy may have some effect on the previously observed fuel constituent migration behavior. Establishing viability of recycled fuel will require that the impact of minor actinides on this phenomenon be assessed. If unacceptable performance is identified, then design improvements must be developed and implemented to ensure viability.

Additional development of fuel design will be directed toward 1) establishing higher operating margins for the fuel at current operating temperatures by increasing the temperatures that lead to deleterious behavior, and 2) establishing the ability to operate at increased outlet temperatures, which might be desired to increase thermal efficiency. Improvements in technology will lead to cladding and duct materials that can endure high-exposure operation, such as the reference HT9, but with improved high-temperature strength over that of HT9. Although this set of activities is important to the long-term utilization of U-Pu-Zr fuel, these issues are not viability issues. Therefore, R&D tasks associated with increased operating margin and increased outlet temperatures are not addressed here.

R&D Tasks

The viability issues described above are best addressed through an irradiation test program. The test program envisioned consists of irradiation of a limited number of recycled fuel assemblies, probably in the JOYO fast reactor in Japan, which will provide performance information to indicate the applicability of the established U-Pu-Zr database to recycled fuel. These irradiations would be conducted at more conservative, nominal conditions initially with remote examinations conducted at interim burnup values. As the irradiation behavior is established, then the more aggressive, 2-sigma conditions would be investigated in fuel assemblies that are designed to operate at or near 2-sigma assembly outlet temperatures. In total, four to six experimental fuel assemblies are envisioned, for irradiation at varying times and rates over a five to six-year period. Additional necessary irradiation tests might be identified during the course of the proposed test program, and specialized post-irradiation tests, conducted in hot cells, are likely. Furthermore, selected irradiated rods of recycled fuel will be tested under transient overpower and undercooling conditions in a facility such as the Transient Reactor Test Facility (TREAT) in the U.S.

3.3.1.4 Modeling and Code Development. The performance of U-Pu-Zr can be adequately predicted using the LIFE-METAL code as developed in the 1990s. However, improvements to the models of phenomena and to the overall code are necessary to support a safety case for recycled fuel. Activities that will be necessary to support a safety case include the following:

- Develop and validate predictive models for the following:
  - Swelling and fission gas release
  - Fuel constituent interdiffusion
  - Fuel-cladding interdiffusion
  - Cladding stress rupture at moderate strain rates and high temperatures.
- Development of a predictive fuel performance code, with the following efforts:
  - Redesign and coding for the base temperature and irradiation exposure model
  - Incorporation of the newly developed models described above.

<u>Technology Gaps</u>

The development of the LIFE-METAL code and its successful use to support the EBR-II Mark-V Driver Fuel Safety Case indicate that U-Pu-Zr fuel behavior can be predicted with correlative models. Therefore, there are no viability issues that require further development of a fuel performance code. However, understanding of the FCCI and fuel constituent phenomena that might be exacerbated in recycled fuel will likely require mechanistic modeling. The incorporation of these models into a fuel performance code can be deferred until a post-viability phase of R&D.

<u>R&D Tasks</u>

- Development of a mechanistic, predictive model for FCCI, which accounts for the enhancing effect of rare earth fission products.
- Development of a mechanistic, predictive model for fuel constituent migration, which incorporates the effect of minor actinides on chemical potentials of different constituents in different phases and on the mobility of the constituents in the fuel.

3.3.2 Mixed Oxide Fuel

Oxide fuel became the reference for most fast reactor programs worldwide in the late 1960s and 1970s. Therefore, a substantial amount of development work with oxide fuel has been performed since that time. Because fast reactors have typically been intended for breeding of additional fissile material from U-238, the fuel compositions considered are primarily $(U,Pu)O_2$, or MOX. A substantial fuel performance database has been established, and burnup capability over 20 at.% has been demonstrated, for example, in the U.S., Russia, and France. Efforts to improve the economics of oxide-fueled, sodium-cooled fast reactors are centered on increasing the outlet temperature (to allow a higher thermal efficiency to be realized) and increasing fuel burnup (to decrease the number of recycle passes per unit energy generated). Of these two objectives, only the materials issues associated with higher outlet temperatures constitute a viability issue.

However, newly envisioned missions for fast reactors raise some additional feasibility issues. In particular, the desire to recycle and consume minor actinides in fast reactors motivates the use of fuels with minor actinides in the as-fabricated condition. Because minor actinide isotopes are typically unstable, they emit substantial radiation fields. Therefore, fuel fabrication with minor actinide-bearing feed must be performed inside shielded enclosures, or hot cells – an additional degree of difficulty over that encountered with MOX fuel fabrication in glovebox-type enclosures, which has been deployed on commercial scale in France and other places. Hot cell fabrication, and equipment maintenance, will be made possible if fabrication processes can be made simpler than those currently deployed in glovebox-type operations. In addition, efficient consumption of minor actinides will require small amounts of actinide loss to secondary waste streams – an important consideration.

Although MOX fuel designs would benefit from continued development and improvement, it is the viability issues stated above that warrant R&D in the Generation IV context. Therefore, viability R&D will be emphasized in this plan, although less urgent issues are discussed as well.

3.3.2.1 Property Determination.

Technology Gaps

After desired compositions of minor actinide-bearing MOX fuel are determined, thermophysical properties and phase equilibria of those compositions must be assessed. It is expected that such properties will not differ much from the better known $(U,Pu)O_2$ compositions; however, safety analyses and licensing bases will require confirmation and reduction in value uncertainties. The following tasks are not considered essential to establish viability.

R&D Tasks

- Measure thermophysical properties (e.g., thermal conductivity, heat capacity, vapor pressure) of minor actinide-bearing MOX at key values of temperature and/or pressure for comparison against $(U,Pu)O_2$ values.
- Evaluate the phase equilibria behavior, including melting temperatures, of minor actinide-bearing MOX.

3.3.2.2 Fabrication Development.

Technology Gaps

Because fabrication of minor actinide-bearing MOX must be undertaken inside shielded hot cells, fabrication processes that are simpler than the currently well-establish pelletizing processes will be highly desirable, or perhaps even necessary. The objectives of new process development will be processes that release less powder contamination, require less equipment maintenance, and lose less actinides to secondary waste streams. Therefore, processes with fewer steps, less powder handling and fewer pieces of equipment will be emphasized.

Candidate process technologies to be considered include vibro-compaction of MOX powder into cladding tubes, such as that being developed and deployed at the Russian Institute of Atomic Reactors (RIAR) near Dimitrovgrad, Russia, and the simplified pelletizing method that is being developed by the Japan Nuclear Cycle Development Institute (JNC). R&D into these processes, or others that might yet be identified is considered important for establishing the viability of an oxide-fueled fast reactor that is deployed to manage actinide inventories.

R&D Tasks

- R&D tasks for fabrication using the Simplified Pelletizing Method:
  - Techniques for adjusting plutonium content during the mixing stage of uranium and plutonium nitrate solutions in a reprocessing plant.
  - Techniques to enhance powder flow techniques, for example by controlling the temperature during calcination/reduction.
  - Pellet-pressing equipment with die-wall lubrication.
  - Pneumatic powder transport systems, including an accountability system for nuclear materials.
- R&D tasks for remote application of the Simplified Pelletizing Method:
  - Remote maintenance.
  - Handling of low-decontaminated TRU fuel, including decay heat removal measures.
  - A turntable type denitration/calcinations/reduction system.
- R&D for vibro-compaction of MOX:
  - Extension of the gelation technique for production of minor actinide-bearing MOX powders.
  - Optimization of gelation conditions.
  - Treatment of gelation waste solution.
  - Optimization of vibro-compaction conditions and parameters for production of dense and uniform fuel column.
  - Development of a nondestructive inspection technique for application to vibro-compacted rods in a remote environment.

3.3.2.3 Irradiation Performance.

Technology Gaps

In general, the irradiation performance of MOX fuel containing additional quantities of minor actinides must be investigated and demonstrated. For example, increased amounts of Am and Cm will lead to increasing amounts of helium generation, which will have implications for fuel performance that are similar to those of fission gas generation. Effects such as this, and others, must be investigated for comparison against the established irradiation performance database for MOX. Irradiation experiments performed by the CEA indicate that additions of Np and Am to standard MOX compositions do not degrade fuel performance substantially. But further irradiation testing, including testing under transient-overpower and loss-of-flow conditions, is required to fully evaluate the performance of the desired compositions. Demonstrating that minor actinide bearing fuel performance is bounded by the known performance of standard MOX compositions (or assessing the behavior and determining the limiting conditions for operation for these fuels) is considered essential for establishing viability.

Increasing coolant outlet temperatures, for the purposes of increasing thermal efficiency, will require the development of a new cladding material (or perhaps the selection of an existing material). Ferritic-martensitic stainless steels, such as HT9 in the U.S., have proven to have excellent resistance, and their use has made possible the attaining of burnups greater than 20 at.%. However, operation with coolant outlet temperatures greater than 500 to 520°C will require an alloy with improved high-temperature strength and improved creep strength. Toward that end, Japan and France have been investigating oxide dispersion-strengthened (ODS) ferritic alloys. These alloys have improved high-temperature strength and creep resistance due to the strengthening mechanisms associated with the oxide dispersions; however, their fabrication and joining (welding) can be difficult. In Japan ODS farritic cladding tubes and joining (welding) technology has been developed. An ODS-clad fuel pin irradiation test is being prepared. Further R&D is required to demonstrate the swelling and creep resistance for high burnup applications in fast reactors, and to improve the fabrication process, welding/joining process etc.

R&D Tasks

- Irradiation testing of selected minor actinide-bearing MOX compositions
- Safety testing of irradiated rods of
- ODS development
- Fabrication studies
- Welding/joining studies
- Irradiation testing.

3.3.2.4 Modeling and Code Development.

Technology Gaps

Current oxide fuel performance codes are well established and functional. The evolution of MOX fuel designs to include minor actinide constituents and advanced cladding materials (such as ODS alloys), however, will require that the codes and models be updated. If fuel performance is not significantly changed, then updating to reflect new fuel compositions should be a simple matter of re-calibrating the models for the new behavior; new phenomena will require the development of new models. Models for the irradiation performance of, the deformation of, and damage accumulation in new cladding materials (such as ODS alloys) must be developed and incorporated into codes. This work will eventually be important to support safety analysis and licensing, but it is not essential to establish viability of the reactor concept.

R&D Tasks

- Evaluate irradiation and safety performance of minor actinide-bearing fuel and update existing models, or develop new models, as appropriate.
- Develop models for behavior of advanced cladding materials and incorporate into fuel performance codes.

3.3.3 Nitride Fuel

Nitride fuel technology is not well established, although work performed to date (mostly in the 1960s through the 1980s) indicates that the fuel type is quite promising. Much of the work to be done includes adaptation of existing technology for oxide and nitride fuel fabrication and design to envisioned applications with increased minor actinide contents. The irradiation performance database with (U,Pu)N is sparse, so initial irradiation testing will investigate potential limits of current designs and identify design improvements. The performance of nitride fuel will benefit from development or identification of improved cladding alloys for high-temperature strength and swelling resistance.

3.3.3.1 Property Determination.

Technology Gaps

The phase equilibria and thermophysical properties of (U,TRU)N compounds are not fully known, although considerable work has been performed in Japan in recent years.

R&D Tasks

- Assessment of phase equilibria of TRU nitrides at temperatures of interest, with particular attention to the affect of minor actinide additions on dissociation temperatures.
- Thermophysical property measurement.
- Confirmation of fuel compatibility and coolant with minor actinide additions

3.3.3.2 Fabrication Development.

Technology Gaps

Pellet fabrication technology for UN and (U,Pu) is fairly well established. However, the adaptation of this technology to (U,TRU)N, perhaps with residual fission products, is not complete. Some work has been performed in Japan and Russia, and new work is begin performed in the U.S. as part of the Advanced Accelerator Applications (AAA) Program, but fabrication with a full range of expected compositions has not been achieved. Other considerations, such as development of techniques that facilitate remote fabrication or to reduce actinide losses to secondary waste streams during fabrication have not been addressed. Of these latter concerns, only the reduction of actinide loss during fabrication is considered a viability issue.

R&D Tasks

- Adaptation of well-established pellet fabrication technology to fuel compositions with minor actinides, with emphasis on suppression of AmN volatility.
- Development of advanced fabrication techniques that reduce or eliminate fabrication losses of actinides.

3.3.3.3 Irradiation Performance.

Technology Gaps

The irradiation performance database for (U,Pu)N is small, but sufficient to indicate that utilization of (U,Pu)N to moderate burnups (about 10 to 12 at.% burnup) may be possible. However, there is little or no experience with (U,TRU)N compositions and no experience that would allow assessment of limiting conditions of operation. An irradiation testing program, similar to that described for Viability R&D with recycled U-Pu-Zr is envisioned.

- Assessment of (U,Pu)N behavior and lifetime potential over the full range of nominal and 2-sigma operating conditions.
- Assessment of transient overpower and loss-of-flow behavior of (U,Pu)N at various stages of burnup, with particular attention to behavior of fuel and retained fission gas and of dissociation effects (if any) during transient overpower events.
- Assessment of the impact of minor actinide contents to the steady-state and transient behavior of (U,Pu)N.
- Development of cladding alloys (or identification and evaluation of existing alloys) with improved high-temperature strength over HT9, but with similar resistance to swelling.

3.3.3.4 Modeling and Code Development.

<u>Technology Gaps</u>

There is no available, validated code for predicting performance of nitride fuel. However, life-limiting phenomena for such fuel at higher burnup values (15 to 20 at.%) have not been identified, and thus have not been modeled. The issues related to viability are limited to the development of models to allow an understanding of life-limiting phenomena. The development of a fuel performance code can be deferred to a post-viability R&D phase, and may be best initiated with the adaptation of a fast reactor oxide fuel code.

<u>R&D Tasks</u>

- Development of mechanistic models for life-limiting phenomena and other performance characteristics as they are identified, or adaptation of those developed for $(U,Pu)O_2$, if appropriate.

3.4 Reactor Technologies

3.4.1 In-Service Inspection and Repair

Preventive maintenance efforts will be emphasized to ensure high plant availability. Improvement of ISI&R technologies are important to confirm the integrity of under-sodium safety related structures and boundaries, and to repair structures in-place quickly.

For this improvement, the system and component design should be carried out taking account of the development targets for the following three elements: (1) high quality sensors under 200°C sodium (sensor technology), (2) accurate remote handling systems such as manipulators that are mobile in narrow spaces (robotics), and (3) high resolution and quick image processing systems (image processing).

3.4.2 Sodium-water Reaction Detection, Mitigation

As for sodium water reactions, it is necessary to enhance the reliability of early detection systems for water leaks. The earlier detection systems, especially against small leaks, would be adopted to prevent the propagation of tube ruptures, and to realize rapid return to plant operation, although the detection sensitivity would be less in the larger steam generators. For safety concerns, wastage data and high temperature creep strength for high chromium steel is necessary. A comprehensive evaluation method for tube rupture propagation behavior would be developed taking account of simultaneous wastage and overheating rupture modes, where the hydrodynamics of sodium and water-steam, the local coolability of water in tubes, etc. would be precisely evaluated. For this purpose, experimental effort would be required.

Multiple failures of mitigation systems could be also significant, even though failures of the blow down system and isolation of feed water are assumed to be beyond the design basis. In such cases, large-scale tube ruptures may occur, and the resulting high pressure may challenge the integrity of the reactor coolant boundary. Experimental data to demonstrate self-limiting behavior of tube ruptures due to sodium movement in the reaction zone is desirable to clarify the design margin for the primary boundary, and also the secondary systems. In providing such a design margin for the boundaries, it is then easy to explain that the chemical potential of sodium never connects to the safety of the reactor.

In addition to the above-mentioned R&D for conventional single tube steam generators, two other approaches might be investigated as alternative designs. The first approach is introducing the double-wall-tube steam generators. This concept minimizes the consequences for sodium-water reactions. Simultaneous penetration breaks (of both tubes) would be beyond the design basis. A second approach is developing a new steam generator without a secondary system, where the possibility of sodium water reaction would be ruled out. The fundamental feasibility of some new concepts, where the lead bismuth alloy is used as intermediate heat transfer medium, is now under investigation.

Throughout these investigations, the most promising steam generator concept from the viewpoint of technical feasibility, construction cost, and consequences of a sodium/water reaction would be selected. These efforts toward more reliable and robust steam generator design can be significant in reducing plant cost and improving plant reliability.

3.4.3 New Materials

Technology Gaps

Development and/or selection of structural materials for components and piping is among the issues of critical significance for development of an economically competitive plant.

12 Cr ferritic steels, instead of austenitic steels, are viewed as promising structural materials for future plant components, because of their superior elevated temperature strength and thermal properties, including high thermal conductivity and low thermal expansion coefficient. With these materials, more compact structural designs are foreseen.

On the other hand, there are some shortcomings to be overcome with these materials. They include degradation of ductility and toughness during high temperature service. Weldability is also a concern. Elevated temperature material strength database should be established for design- by-analysis purposes.

R&D Tasks

- Accumulation of material strength database with a focus on the creep-fatigue strength in the fast reactor plant temperature range.
- Improvement of toughness and ductility of the material.
- Welding procedure development optimized for this class of materials.
- Elevated temperature strength data of welded joints.
- Manufacturing technology development for thick plate materials and thin walled heat transfer tube materials.

3.4.4 Concept Design

At present this is a placeholder for reactor concept design activities. Since the final product of Gen IV R&D is by the established ground rules, a completed conceptual design, these design activities can hardly be ignored. Moreover, design activities (analyses and trade-off studies) provide valuable information to guide the R&D in the base technology areas. The TWG 3 recommended approach is to carry a lightly funded design activity throughout L1/L2 R&D, of course accelerating it to high-priority status when appropriate.

3.4.5 Nuclear Security

See Section 3.1.4. This is a placeholder until the issue clarifies. It will be important to resolve prior to initiation of conceptual design.

4. R&D PLAN FOR CONCEPT SETS L6/L4

4.1 Identification of Technological Gaps, High Potential Payoffs, and R&D Requirements

4.1.1 Focusing R&D in a Science-Based Program

The experience base with lead and lead-bismuth eutectic (LBE) coolant in fast reactors is not nearly as robust as for sodium coolant. Therefore the L6/L4 concept sets for lead or LBE cooled fast reactor concepts will require several years of "science-based" R&D. This work will establish some of the basic data and understanding required for moving beyond preconceptual design.

The required science-based R&D lies in the following areas:

<u>Fuels</u>

1. Choosing a fuel/clad/coolant combination which is feasible on the basis of chemical/thermal/structural compatibility including basic data and integral testing
2. Devising a recycle/refabrication/waste form strategy, and developing basic chemistry at a bench scale

<u>Heat Removal</u>

3. Developing high quality basic thermo-physical properties for the coolant
4. Developing needed thermo-physical heat transfer and pressure drop correlation data required to do core heat transport designs
5. Developing coolant chemistry control technologies and polonium control measures

<u>Core Design</u>

6. Developing reliable basic neutronics data for lead or LBE-cooled lattices with representative fuel/clad/coolant choices. Lead, bismuth, and minor actinide data are areas of important technology gaps
7. Developing an overarching safety strategy

<u>Structures/Components</u>

8. Developing preconceptual design strategies for reactor structures, support structures, and refueling, accounting for the density of coolant exceeding that of components
9. Developing design strategies for ISI and repair of the critical components.
10. Developing basic phenomenological data required for heat transport component designs
11. Developing the knowledge of corrosion/erosion mechanism for structural materials immersed in lead or LBE. Consequent selection of candidates materials and/or protective coatings.
12. Developing high temperature code cases for the new materials

<u>Energy Converters</u>

13. Developing the energy conversion technologies to exploit recent evolution in energy converters, to service new markets, and to reduce costs

14. Developing the coupling technologies between the nuclear heat source and the energy conversion, and the safety strategy for such coupling.

The listing above defines an extensive scope for a science based R&D program. It will be essential to focus the R&D on special opportunities, and possible showstoppers, of these reactor and related fuel cycle concepts. In a predominantly science based R&D effort, keeping in mind special opportunities and possible showstoppers provides the discipline to focus the effort by ranking the R&D activities according to how relevant they are with regard to filling crucial technology gaps identified as essential for exploiting the potential payoffs. That has been accomplished by the TWG – as outlined in the next three sections.

4.1.2 Targeted Missions and Expected Payoffs of the L6/L4 Concepts

Mission of L4 Concepts

The L4 concepts include four concepts from the INEEL/MIT team:

M19 An LWR Spent Fuel TRU Burner

M23 An LWR Spent Fuel Minor Actinide Burner

M18 An LWR TRU Burner with Direct Contact Steam Generation

M27 An LWR TRU Burner using a pebble bed core design

All four concepts are targeted to the same mission in a synergistic energy park—that of cleaning up the spent fuel discharged from LWRs by interposing a closed multi-recycle fast spectrum reactor between the LWRs and the repository. LWR spent fuel is partitioned into three product streams: the U is set aside, the fission products go to waste and the TRU are recycled into the L4 reactors for fission. All four concepts are targeted to perform this waste management function in a cost effective fashion by using the fission heat for generating electricity for revenue.

The fuel used in the four L4 concepts is a proposed new metallic fuel of composition Th/U/Pu/MA/Zr (M19, 18, 27) or Th/U/MA/Zr (M23). The thorium is included mainly to reduce the U238 that is normally present. When present, however, the U238 breeds plutonium. Here the U238 amount is limited to that needed to denature the U233 that is bred from thorium. Together the thorium and the U238 provide a sufficient Doppler reactivity coefficient, and the material bred (largely U233) serves to reduce burnup reactivity loss. This then increases the reactor cycle length (an economic advantage), or it reduces the reactivity initially invested in control rods (a safety advantage).

The L4 concepts have the flexibility to run in a fissile self-sufficient mode. They are not suited for net fissile production with short doubling time. In this regard, L4 shares attributes similar to concept set L5, the large (and largely Russian) lead-cooled reactors with either advanced aqueous or dry processing. Concept set L5 was not recommended for further consideration, although clearly the technology embodied there would benefit from results of the science-based program.

The power density of these concepts is relatively low compared to the L1/L2 sodium cooled systems. This is mainly due to the lower heat transfer coefficient, and is further reduced by the limitation on the coolant speed to avoid excessive corrosion/erosion of the structural materials. The specific fissile inventory is relatively high; and the achievable doubling time is relatively long. That is why these concepts are generally targeted for a fissile self-sufficient deployment strategy. Two sources of initial working inventory of fissile can be foreseen: enriched uranium from virgin ore, or existing stores of transuranics recovered from discharged thermal reactor once-through cycles.

Potential Payoff

These concepts are proposed as a more cost effective alternative to Accelerator Driven Systems (ADS) which, while not a part of Gen IV, are under extensive study worldwide and are targeted at the same mission: i.e., LWR spent fuel management. Like the ADS systems, they use a fuel which contains little U238 (none in the ADS case) – and thereby attain a maximum net destruction of TRU per unit of heat generation.

Both reactors (lead-based or sodium cooled) systems intend to outperform the ADS on cost effectiveness by:

- Avoiding the capital and operating cost of an accelerator
- Avoiding the need to divert part of the generated electricity from sale back to hotel load to run the accelerator
- Avoiding the frequent interruption of power generation that results from accelerator trips—and which, because of supply unreliability, reduces the market value of the generated electricity.

Additionally, both reactor types are intended to outperform the ADS on safety performance by achieving passive safety response to ATWS initiators, and by employing an innate HCDA quenching mechanism

Mission of L6 Concepts

The L6 concepts are all small "battery" type concepts, in the 50 to 130 MWe range, designed for long refueling interval (15-20 years), and cassette or entire module refueling. Recycle is supported by a regionalized fuel cycle service. These systems are fissile self sufficient, passively safe power plants, with passive load following capability, which both simplifies the reactor design and allows for a non-safety-grade balance of plant.

Potential Payoffs

These plants are targeted for either of two client bases which are foreseen to grow dramatically in the first half of the twenty-first century. First are developing countries which may not wish to deploy indigenous fuel cycle infrastructures but still desire the energy supply security and emissions-free benefits of nuclear energy. The small size is matched to small grid size and to scarcity of capital for economic development projects. A factory-fabricated, transportable turnkey plant allows for short intervals between raising capital and generating product and revenue. The passive safety/passive load following feature permits a non-safety grade balance of plant and civil works for the NSSS—features which allow for indigenous supply sources and job creation for the conventional (non-nuclear) elements of power plant construction and subsequent operation and maintenance.

The second category of client is merchant plants (Independent Power Producers) in developed countries. Such clients must pay merchant rates for capital, match capacity addition closely to demand growth, and shorten the period between raising capital and generation of revenue.

These two markets will have to grow in order to justify the investment that must be made to produce L6 reactors and attendant recycle facilities. Large factories would be needed to mass-produce standardized plants in volumes sufficient that economy of mass production will compensate for loss of economy of scale.

The lead-based coolants are ideally suited for battery type plants of long refueling interval. The power density must be derated in any case, so that a fuel of given burnup limit will last many years before it reaches its burnup limit. Thus the lattice can be opened to increase coolant volume fraction without neutronic penalty, because unlike the case for sodium coolant, the lead-based coolants have extremely small neutron absorption and neutron slowing down properties. Given a low power density and high coolant volume-fraction lattice, natural circulation will remove heat at full power. The excellent neutron reflection properties of lead or LBE coolant surrounding the core, and the high-energy neutron spectrum that results, permit fissile self-regeneration in the core lattice itself (i.e., breeding blankets are unnecessary). This permits near-zero burnup reactivity loss over the full 15 to 20% burnup interval—which is the key to enabling passive load following and passive safety. This in turn enables coupling the reactor heat transport system to a balance of plant having no safety function. The latter can then be produced indigenously within the country of deployment.

This matching of innate properties of the coolant with the design needs of this particular product cannot be matched using water, gas, or sodium as coolant; it is the natural niche where the properties of the lead-based coolant outperform all other choices.

4.1.3 Implications of the Missions on Design Innovations

A salient characteristic of the L4 concepts is their relaxed requirements for breeding. Fast spectrum systems have an inherent flexibility with regard to breeding/incinerating. However, the L4 mission permits to draw on basic design features (e.g., medium size power ratings, moderate power densities, high density fuels, lack of breeding zones) benefiting from relaxed requirements for achieving short fissile doubling time. Various special opportunities follow from this, such as the possibility of designing long-life cores. More important, perhaps, is the potential for actinide waste management (transmutation), while at the same time, achieving enhanced safety performance.

The potential for transmutation is a direct consequence of the relaxed requirement for neutron economy. Since excess neutrons are not needed for breeding they become available for incineration, not only of minor actinides but also the transmutation of long-lived fission products. This special opportunity is due to the high energy neutron spectrum, but also derives from the chosen nitride or metallic fuel types (capable of high concentration of minor actinides, or even totally “dedicated”, i.e., fertile-free fuels), and from the chosen coolant, all of which enhance the transmutation potential.

Characteristics that enable passive safety include: (1) high lead-based boiling temperature, so that local boiling is practically precluded, (2) negative coolant-void reactivity coefficient (mainly due to the low moderating power of lead-based coolants and to the lack of fertile blankets), (3) enhanced natural circulation, (4) large negative temperature feedback coefficients, and (5) low excess reactivity margins.

The L6 concepts also rely on a relaxed fissile doubling time requirement. As in the case of the L4 concepts, this, along with the coolant characteristics and the ensuing thermal hydraulics design, leads to long-life core concepts. For L6 concepts, this is the defining characteristic: these are “battery” (or “cartridge”) type nuclear reactors, delivered turnkey to the site, with no on-site refuelling or fuel handling. L6 concepts share many of the characteristics described above for the L4 concepts; therefore, many of the special opportunities noted for the L4 concepts also apply for L6. Moreover, L6 concepts are generally characterized by a high degree of innovation not only in design but also in overall supply and support infrastructure targeted to achieve economic competitiveness in spite of small power rating.

Finally, and perhaps most important regarding payoffs, the L6/L4 concepts offer special opportunities in high temperature applications, including coupling to a Brayton cycle; hydrogen production via water cracking; and process heat missions. This is because the coolant boiling temperature is very high and offers potential to reach temperatures unavailable to sodium, at least sodium at ambient pressures.

4.1.4 A Process for Linking Potential Payoffs Technology Gaps and Potential Showstoppers with R&D Needs

Prior to developing the R&D plan, the TWG followed a process to systematically identify:

- Potential major payoffs for innovative features of each specific concept in the L6/L4 sets
- Technology gaps that must be filled in order that the concept meet its potential
- Those gaps (if any) which are potential showstoppers—capable of preventing the concept from meeting its potential.

This identification process was organized by “function to be performed” for the power plant and fuel cycle concept (these functions are shown as column headings in Table 10). The process was then extended to link the gaps with the R&D which would be required to fill the gap. Since the concepts are at a stage of early development, they first require fundamental data to be developed in order to support later Title I design. Therefore the R&D was organized by technical discipline, as indicated as rows in Table 10.[c]

After completion of the identification process, which was done via extended discussions at two of the quarterly TWG meetings, Table 10 served to document a (Payoff/Gap/Showstopper/R&D needs) profile for the concepts in L6/L4 and highlighted the information required for four activities:

Screening out concepts for which the payoffs were small versus the extent of gaps and/or the potential for showstoppers. This included concepts M27 and M18, and as mentioned the Russian-dominated concept set L5. However, some of the innovations within these concepts or sets were deemed worthy of development, but for alternative concept designs in the future.

Identifying the concepts for which the payoffs were highest versus the extent of gaps and/or the potential for showstoppers.

Identifying crosscutting R&D for high payoff technologies which supported multiple concepts or which would otherwise be missed because the only concept using it was screened out.

Organizing the time sequencing of the R&D activities to fill the gaps for the remaining promising concepts

Table 10 summarizes the outcome of this process.

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c. For a Title I/Title II development project such as would be followed for more mature technologies the project structure itself will integrate the development tasks among participants. Here, for

Table 10. Payoffs, technology gaps, potential showstoppers, and required R&D for concept Sets L6/L4.

Function / Discipline	Compatible Fuel/Clad/Coolant	Maintain Neutron Balance	Reactor Control	Heat Removal from Lattice	Heat Transfer to Components and BOP	Reactor Structures, Shielding and Refueling	Overall Safety Strategy	O&M Strategy	Fabricability and Costs	Fuel Cycle (Recycle, Refab and Waste Forms)	Energy Conversions
Fuel	"+" high temperature heat; coolant chemically inert; high natural convection potential (passive safety) "-" corrosion; Pb freeze "=" fuels, coolant/clad/fuel compatibility (high temperature materials) -New Materials Coatings Screening for High Temp	"+" total negative void coefficient transmutation; open lattice; natural convection; long-life core "-" nuclear data (Pb, Bi)					"+" less severe recriticality			Common to LMTWG concepts ·Nitride ·Dispersion Fuel	
Materials and Coolant				"-" autocatalytic redistribution of flow in the open lattice when heat transfer coeff decreases "-" heat transfer sensitive to coolant contamination	"+" eliminate intermediate heat exchanger "-" higher melting temperature for Pb	"+" simpler refueling; less demanding atmosphere control "-" Po (and other) radioactive hazards	"+" no fire; "-" Po, Pb hazards	"+" no fire; "-" Po. Pb hazards	"="high temperature materials		
Neutronics											
Thermal Hydraulics											
Structural											
Instrument-ation and Diagnostics	"-" ISI										
Economics	-Fluence limits on high burnup		+Passive load Follow				-Safety interaction with BOP for non electric products		+Factory Fab =Advanced Materials		+Brayton (CO_2) +Process Ht

Legend

+ denotes high potential payoff

- denotes technology gaps

= denotes potential showstopper

4.2 List of Technologies Having High Potential Payoff

The lead-based L6/L4 concept sets, like the sodium cooled L1/L2 concepts employ closed fuel cycles based on advanced recycle/refab technologies which use the “dirty fuel/clean waste” design philosophy. Under this philosophy all actinides are multi- recycled to total fission consumption and only fission products (and trace losses of actinides) go to the repository. The full energy potential of the actinide resource is harvested and the waste requires only ~300 years of sequestration until it has decayed to a toxicity level equivalent to the original ore. The fuel maintains a spent fuel standard of radioactivity self protection at every step of the fuel cycle; it is of an isotopic mix that is unattractive for weapons use; and only trace amounts of fissile material ends up in geologic storage. This is a high potential payoff technology.

In fuel cycle innovations, the L6/L4 concepts are not distinguishable from those of L1/L2 although the specific fuels under consideration are different. However, it is for the power plants where the L6/L4 concepts present a remarkable range of innovations in fast reactor plant design which distinguish them from the more mature concept sets, L1/L2. While some of these innovations are based on exploitation of the relatively inert chemical properties of lead or LBE, other innovations are based on coupling to modern energy converters, producing non-electric energy products thru higher temperatures, and responding to emerging markets (e.g., deregulated markets and new markets in developing countries). This is all accompanied by innovative manufacturing, delivery and fuel cycle support strategies (e.g., the nuclear battery concepts).

Table 11 is a list of innovations presented in one or more L6/L4 concepts. Several of these innovations are highlighted in the discussions below. Even if the specific concept which employs it did not receive a high rating for potential, the innovation itself holds a high potential payoff and should be pursued in the Gen IV science-based R&D plan for concept sets L6/L4.

Supercritical CO_2 Brayton Cycle

The L4 concepts include a supercritical CO_2 Brayton cycle energy converter as an option. This energy converter concept would seem to hold remarkable potential for replacing the Rankine steam cycle to which fast reactors have traditionally been coupled.

The supercritical CO_2 Brayton cycle (operating at 550°C and 20 MPa as proposed) brings three favorable features. First, an efficiency of 45 to 47% can be attained, at (only) a 550°C reactor coolant outlet temperature. Second, the astonishingly smaller footprint and the simplicity (many fewer components) of the supercritical CO_2 Brayton cycle as compared to the Rankine steam cycle holds potential for significant capital cost reduction in the BOP.[d] Third, the simplicity of the Brayton cycle allows for significantly reduced O&M staffing in the BOP component of operating expense.

A substantial R&D program would be required to develop the supercritical CO_2 Brayton cycle. It was proposed in the 1960s but was only partially developed at that time. Much of the technology is in

d. BOP contribution to standard fast reactor plants is in the range of 1/3 to 1/2 of overnight cost.

Table 11. Innovations found in L6/L4 concept sets.

Fuel Cycle/Logistics	Energy Converters	Heat Transport	Plant Control (Passive Load Following)	Fuel Types	Recycle Types
Regional fuel cycle centers for countries which don't want an indigenous fuel cycle	Supercritical CO2 Brayton cycle	Direct contact heat exchange	Pebble bed core with semi-continuous refueling	U/TRU/Zr metal alloy	Metal Pyro □ Nitride
Long life cassette refueling	Supercritical Rankine cycle	Direct contact steam generation	Zero burnup control swing, internal conversion ratio of one, i.e., fissile self-sufficient core without external blankets	Th/U/TRU/Zr metal alloy	Advanced aqueous process
Transportable turnkey plants	Thermochemical water cracking	Lift pump		Th/U/MA/Zr metal alloy	AIROX
	Desalination bottom cycle	Natural circulation at full power (open lattice, low power density)		U Nitride	
	Chemical heat pump	Copper or liquid metal bonded steam generator design (eliminate secondary loop in sodium systems)		U-TRU Nitride	
				U Oxide	
				U-TRU Oxide	

hand, as an extensive industrial experience base exists for supercritical CO_2 pumping and handling in the petroleum industry, where CO_2 is compressed to supercritical pressures, piped for many miles, and injected into depleted oil fields to simultaneously drive out the last of the petroleum and to sequester the CO_2. Additionally the British AGR reactors were CO_2 cooled; so substantial materials compatibility experience at somewhat higher temperatures and much lower pressures (650°C and 4.2 MPa) already exists. Turbine and recuperator designs do not yet exist, however.

Reactors for Process Heat and Hydrogen

The extremely high boiling temperature of lead at atmospheric pressure suggests its application in fast reactors for process heat applications and specifically for hydrogen production by thermochemical water cracking cycles. The manufacture of hydrogen from water and nuclear heat provides a pathway to "greenhouse gas emission-free" energy sustainability by opening a route for nuclear to contribute to the nonelectric energy sectors which are currently served only by fossil fuels.

The R&D issues here center on structural materials compatible with lead-based coolants in a radiation environment in the 750 to 850°C temperature range. The further constraints are to avoid neutron moderation by structural material and to provide for long life and affordable fabricability of the high temperature structures.

Beyond the reactor structural materials issues, R&D is necessary to carry the thermochemical water cracking cycles from the bench to the commercial scale and for developing suitable safety strategies for coupled nuclear/chemical plants.

Desalination Bottoming Cycles

Water is forecast to become a saleable commodity in high demand during the first quarter of the 21st century. The reject heat from a heat engine power plant (which is currently wasted) could easily be diverted to a desalination bottoming cycle at plants located near seawater or brackish water supplies. In a deregulated market, this provides for a storable energy product and an additional revenue source.

Such bottoming cycle plants use "off the shelf" technology. and an experience base exists already at numerous LWRs deployed in Japan. Very little R&D is needed here; economic and market analyses are required.

Lift Pumps

Lift pumps have been used in several L6/L4 concepts; in one case they are combined with direct contact steam generation; in another case cover gas is recirculated.

The lift pump works by injecting gas into the coolant riser above the core outlet in a pool plant layout. The resulting decrease in effective density of coolant in the upcomer chimney inside the core barrel compared to normal density in the downcomer inside and below the invessel heat exchanger impels the primary coolant through the core and heat exchangers. Thus the primary pump is replaced with a simpler and more easily maintained blower, which can be located outside the vessel. The potential for simplification and cost reduction is high and especially so for concepts which envision very high temperature coolant—because the blower can be operated out of vessel at moderate temperature.

R&D would be required to assure a bubbly flow regime in the upcomer, adequate gas/liquid separation as in BWRs to avoid bubble carryover into the downcomer for collection below the grid plate and release into the core, and gas blowers which can accommodate cover gas with trace amounts of coolant.

Direct Contact Heat Exchange/and/or Steam Generation

Gas bubbled through heated lead is heated without temperature drop across heat exchanger tubes. It receives its heat addition at the high temperature of the coolant (a benefit in approaching theoretical Carnot cycle efficiency), and (with proper geometry) can simultaneously function as a lift pump for impelling the coolant around a heat transport loop. Even feedwater can be injected into hot coolant without a vapor explosion if the pressure is maintained above a threshold for Taylor instabilities at the coolant/water interface.

This idea is worthy of R&D for application in secondary circuits of the heat transport path—allowing the reactor to maintain the ambient pressure which is a hallmark of the safety case for liquid metal cooled fast reactors. Direct contact of He or CO_2 with lead-based coolants may be possible; with sodium cooling only helium is. The huge potential payoff in eliminating the capital, replacement and maintenance costs of steam generators and heat exchangers should motivate a significant R&D effort.

R&D has been carried out on this idea for many years at the Ben Gurion University of the Negev (Branover, Lesin) and the technology is closely tied to the many years of ongoing research on vapor explosions carried out at the University of Wisconsin and at Argonne National Laboratory. R&D on phase separation is required, being a key feasibility issue when the working fluid gas is going into rotating machinery. On the other hand, when the gas will be going into process heat chemical plant applications, chemical compatibility of the chemical reagents with trace amounts of the lead may be a less restrictive constraint.

Close Coupled Heat Transport Loops

Intermediate heat transport loops are indicated when there is a chemical incompatibility of the reactor primary coolant with the BOP working fluid, (e.g., sodium and steam), or when a need exists to provide multiple lines of defense between radioactive primary coolant and the energy product (e.g., district heating grids, desalination). Such intermediate heat transport circuits, while important to the safety function of the plant, significantly add to capital and operating costs.

The TWG 3 concepts sets included several proposed means to retain the safety function at reduced cost by "close-coupling" the primary and BOP fluids across a low thermal resistance heat transfer medium. Two examples illustrate the notion. In one the primary coolant-containing tubes are interspersed with tubes containing the BOP working fluid in a close-packed bundle containing solid copper as an interstitial heat transport medium between the tubes. In another design the copper function is performed by a liquid metal (such as LBE) which is compatible with both primary and BOP fluids.

Whereas the direct contact heat exchange idea holds potential to reduce costs when primary and BOP fluids can be physically contacted and are at the same pressure, the close-coupled heat transport idea can potentially save cost when that is not the case. Both technologies are high potential payoff and deserve R&D attention in the Gen IV program.

Passive Load Following

A passive safety strategy is employed in all liquid metal cooled concepts in the L1, L2, L6 and L4 concept sets. It is based on the use of reactivity feedbacks to provide inherent adjustment of the power level to match the heat removal rate in upset conditions. The heat removal rate may be adversely affected by upsets in pumping power, cold leg temperature of the heat transport loop, or reactivity control. When such an off-normal event leads to a coolant/fuel temperature rise, innate thermostructural reactivity feedbacks can be designed in a way to reduce reactivity, which causes power level to decrease, which brings temperature back into line.

Many of the L6 concepts take this one step further—to employ feedbacks as the only reactivity controller and to thereby eliminate the active reactivity controller—achieving a passive load following strategy in addition to passive safety. (In some cases even the external control of pumping power is eliminated and only natural circulation is employed.)[e]

This design approach holds the potential to further simplify plant design and to make the reactor safety response innately self protecting no matter what events occur in the balance of plant or control room—including incorrect human actions and/or maintenance errors.

R&D on this strategy started during the latter days of the IFR project, but was not completed. Its potential payoff warrants resumption as part of the Gen IV R&D effort on concept set L6.

4.3 List of Top-Ranked Technology Gaps Requiring Resolution During the Generation IV Viability R&D Campaign

The viability R&D program for the lead-based cooled concept sets, L6/L4, will be "science-based" (rather than concept-specific) owing to the early stage of technology for these coolants in the GIF countries.[f] This "science based" viability R&D will be directed primarily to multiple facets of two key areas; materials and fuel cycle.

First, it is necessary to establish a suitable level of knowledge and understanding of key phenomenological behavior of materials (coolant/cladding/fuels) and structures for high temperature service in the ranges contemplated for Generation-4 commercial power plants. The knowledge must achieve a level needed to establish technical viability of the L6/L4 concepts overall (a level which has largely been attained already for the sodium based concept sets L1/L2).

Second the viability R&D program for concept sets L6/L4 must address the recycle/waste form production/refabrication and fuel irradiation performance of those fuel compositions which are not shared with the L1/L2 concept sets—specifically U/TRU mixed nitride fuel and thorium-based TRU and minor actinide fuels.

The areas of top-ranked knowledge gaps to be addressed during the viability R&D phase of Gen IV are enumerated in Table 12 for concept sets L6/L4. These are the top prioritization areas culled out from the much more complete discussions of technology gaps and needed R&D contained in Section 4.4; they are organized in Table 12 under the broad categories of:

- Fuel/Clad/Coolant performance
- Reactor technology
- Safety, neutronics, and control
- Fuel cycle.

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e. Active safety systems and safety rods are retained in the design, of course.

f. The Russian technology base in LBE cooled reactors benefits from their alpha-class submarine experience. Besides being incompletely diffused into GIF country experience, the temperature range, neutron spectrum, choice of fuel and application of the Russian experience are different from Gen IV targets.

Table 12. Top ranked technology gaps for concept Sets L6/L4—(to be resolved during the viability R&D program).

I. Fuel/Clad/Coolant Performance

A. Issues

- Initial service condition is at 550°C in LBE
  - Gen IV longer term potential is for 800°C in lead
- Transition concepts employ U-oxide and MOX fuel

Gen IV Longer-Term Concepts employ

U-TRU-nitride

Th-U-TRU-Zr Alloy

Th-U-MA-Zr Alloy

B. Salient Gaps in Technology Requiring Viability R&D

- Thermo/physical/chemical properties of fuel (as a function of fabrication method)
- All aspects of steady-state irradiation performance of a given (coolant/clad/fuel) pin design
- Transient performance of a given (coolant/clad/fuel) pin design unirradiated and irradiated pins TOP, LOF, and LOHS conditions

II. Reactor Technology

A. Issues

- For 550°C service conditions in LBE it is known that austenitic steels in contact with cold pool temperatures (~400°C) are established; hot pool service conditions may require further checking
- For 800ºC service conditions in lead, a totally new structural material selection is necessary
- The high density of the lead coolants leads to unfamiliar challenges in reactor layout and support to account for:
  - Internal structures will float in the coolant (effective reversal of gravity)
  - The vessel itself will contain immense weight (affecting support approaches)
  - High temperate concrete will be needed around any primary system operating at 800ºC temperatures.

B. Salient Gaps in technology Requiring Viability R&D

- Structural materials for primary system components
- Inservice inspection and repair
  - Under-coolant viewing (and for example, inspection robots)
  - Ultrasonic testing in lead or lead-bismuth (these are issues shared with concept sets L1/L2; however the service conditions differ in both temperature and in density of the coolant)

III. Safety, Neutronics, and Control

A. Issues

- The L6/L4 concept sets share with L1/L2 the need for improved basic nuclear data for minor actinides
  - Moreover, basic nuclear data for Pb and Bi are currently in poor shape as well
- The L6/L4 concept sets employ passive safety responses to accident initiators; the L6 concepts go beyond that to employ a passive load following control strategy. Thermo/structural feedbacks are key to these strategies.
- The phenomenology of new (coolant/clad/fuel) combinations under severe accident conditions is currently unknown

B. Salient Technology Gaps Requiring Viability R&D

- Basic nuclear data (neutron interaction cross sections) of Pb and Bi in the neutron energy range from thermal to 10 Mev

Table 12. (continued).

- Operational and DBA safety
  - Local flow blockage and response in open lattice, ductless designs
  - Passive feedbacks in open lattice, ductless designs
- Severe accident safety
  - Rationalization of HCDA phenomena/analyses/approach (for example nitride volatilization)

IV. Fuel Cycle

A. Issues

- The L6/L4 concept sets all use closed, full actinide recycle fuel cycles based on either advanced aqueous or on dry technologies. They employ remote refabrication and customized fission product containing waste forms
- Because the candidate fuels are different from those used in the L1/L2 concept sets, supplemental (i.e., incremental to L1/L2 targeted R&D) must be undertaken

B. Salient Technology Gaps Requiring Viability R&D

- Irrespective of fuel cycle chosen, if nitride fuel is used, it may be an economic imperative to recover and recycle N-15
- If the pyroprocess is used, all of the gaps noted for concept sets L1/L2 are still present, plus the additional steps of reconversion to nitride and remote fabrication

4.4 R&D Plans for the L6/L4 Concept Sets

As shown in Table 10, the R&D plan for L6/L4 concept sets is organized by function required to be performed in the system concept—e.g., achieving compatibility of fuel/coolant/clad, maintaining neutron balance, etc.; these are the columns in Table 10. Because the R&D for concept sets L6/L4 will be "science based" rather than "specific concept based" during the viability R&D phase, the R&D effort for each function is further subdivided into technical disciplines to account for the disbursement of the science-based R&D to technology-specific investigators. The R&D plans are discussed next in the order of the columns of Table 10.

4.4.1 Fuel/Clad/Coolant Performance R&D

Technology Gaps

The thermo/chemical/structural compatibility of the fuel/clad/coolant combination for the L6/L4 concept sets is one of the crucial viability issues to be addressed during the viability R&D campaign. Totally new combinations of coolant and fuel are proposed for these concepts, and the clad must be compatible with both.

The fuel candidates for concept set L6 include TRU Nitride, and U/TRU/Zr metal alloy. The candidates for concept set L4 include Th/U/TRU/Zr alloy and Th/MA/Zr alloy (an alternate form might be a met/met dispersion fuel of actinide/Zr alloy particles dispersed in a Zr matrix).

At the coolant/cladding interface lie the much discussed issues of coolant chemistry control and choice of ferritic-martensitic steel of an appropriate chrome and silicon content and appropriate grain structure needed to prevent clad dissolution, intergranular corrosion, and oxide layer spalling. However, the range of temperatures (up to 800°C) and coolants (Pb vs Pb-Bi) contemplated for the L6/L4 concept sets requires R&D which goes well beyond that which would merely transfer Russian submarine experience to the West. In particular, service at 800ºC in Pb will call for altogether new cladding materials—perhaps refractories; perhaps special coatings; perhaps ceramics—and these materials may facilitate an entirely different approach to coolant chemistry control than is the traditional oxygen partial pressure approach of the past. Moreover, whatever the new materials are, they must meet numerous other requirements of fabricability, radiation tolerance (low swelling and irradiation induced creep), strength—and compatibility with the new fuel types targeted for the specific missions for the L6/L4 concepts. A fresh look at fabricability, low cost, and nontraditional (aerospace) manufacturing technology may be applied.

At the fuel/clad interface, supplementary to chemical compatibility are the multitude of issues regarding fuel swelling and fuel/clad mechanical interaction; choice of bond material; fuel restructuring; component redistribution and fuel/clad chemical interaction; fuel thermal conductivity and diffusivity versus burnup; axial growth, etc., etc.

The fuel, the bond material, and the coolant should all be chemically compatible to allow for run beyond cladding breach.

The fuel must be remotely fabricable; its fabrication process must not lead to costly rework nor to loss of volatile actinides nor loss of actinides into a waste stream, and so on.

In short, a major fuel pin development campaign for totally new fuel/clad/coolant combinations at new service temperatures is to be undertaken for the L6/L4 concept sets, and in particular technical viability of at least one of the proposed candidate combinations must be established during the viability R&D phase of Gen IV.

R&D Plan

The R&D plan has to deal with three developments used in various L6/L4 concepts:

(a) Technology confirmation and extension to Gen IV of Russian experience for Pb-Bi at ~550°C with fuels already known to be compatible with Ferritic Martensitic Steel

(b) Introduction of new fuel types TRU-Nitride and ThUTRUZr alloy into the Pb-Bi Ferritic Martensitic Steel system at conditions of ~550°C.

(c) Totally new systems of Pb coolant, nitride fuel, and new cladding material at 800°C.

It is necessary in the Viability R&D phase to complete (a) and to take (b) and (c) to a point where feasibility has been established.

Technology Confirmation and Extension to Gen IV of Russian Experience

Lead and lead-bismuth eutectic (from here on simply called lead-alloys) are used as coolant in these advanced reactor system concepts. Although there is a significant development and deployment experience base in Russia, lead-alloy nuclear coolant technology is not at the same technological readiness level (TRL) outside Russia. Its TRL is also considerably lower than that for sodium coolant.

Recent development of lead-alloy spallation target and coolant technology worldwide for accelerator driven systems (ADS) has advanced the state of the art in the West considerably. There is now substantial amount of experimental evidence that the main features of the Russian lead-bismuth eutectic (LBE) nuclear coolant technology are valid for forced circulation small to medium loop type systems. Corrosion tests by various international groups indicate that there are qualified structural materials (US, European and Japanese) for the temperature and flow conditions of the Russian reactors.

However, to achieve the high potential aimed for in the advance reactor system concepts, a significant amount of R&D is needed in the areas of materials and coolant chemistry control. The following subsections will evaluate the technological gaps given the current status, and outline the R&D tasks that will bring the technology to levels suitable for engineering scaled demonstration, before designs for demonstration plants are feasible.

Technological Gaps and R&D Tasks

Reference Coolant Technology and Materials

Russian LBE nuclear coolant technology relies on active control of the oxygen thermodynamic activity in LBE to reduce corrosion and coolant contamination. Within this framework, a series of structural materials were developed and tested for enhanced corrosion resistance and acceptable lifetime. The operating temperature is below 550°C, with fuel cladding temperature below 650°C. The LBE flow velocity design limit is about 2 m/s to prevent the onset of erosion. This is being validated in international R&D for ADS development worldwide (e.g., AAA, FP5, 6, LIPSOR) and will serve as the reference technology for future development.

The oxygen control technique, when properly applied, leads to the formation of "self-healing" protective oxide films on the surfaces of the materials in contact with lead-alloys. This is because the base element (typically Fe) and alloying elements (Cr, Ni) of many structural materials have higher chemical affinity to oxygen. Without such protective measures, Fe, Cr and especially Ni all have non-negligible solubility in lead-alloys that causes severe dissolution attacks.

Oxygen sensors and control systems are thus the core components of the reference coolant technology. Alloying materials with elements promoting tenacious and protective oxides (e.g., Si and Al), or treating/coating the surface with appropriate materials for enhanced corrosion resistance, have been developed and tested with oxygen control.

The gaps are assessed on available Russian data and development in ADS communities worldwide. The R&D tasks are oriented toward validation of key Russian technology components, e.g., oxygen sensing and control, testing of current "nuclear grade" materials, and investigation of the scientific underpinning.

Oxygen Sensors and Control System

Gap: Reliable, calibrated oxygen sensors that can withstand the demanding conditions in reactors with sufficient lifetime are not yet available outside Russia. It is especially lacking in terms of radiation resistance testing. The efficiency and reliability of the oxygen control systems, with gas and solid mass exchange options, are being tested and need further development. Methods of oxygen control need to be selected and optimized (H_2O and H feed; CO_2/CO feed; solid oxide dissolution; electrochemical O_2 production are all candidates). Also assurance of proper mixing throughout the entire melt must be provided.

R&D: This is the critical area for coordinated development and improvement of oxygen sensors and control systems. Without adequate mastering of this technology, most other experiments (materials and thermal hydraulics) will not have sufficient coolant chemistry control to be meaningful. The performance of oxygen control technique in the 350–650°C flowing LBE, the sensor response and stability, robustness including resistance to thermal cycling, sensor calibration, need to be investigated thoroughly in static and dynamic environment. The promising sensor systems should be irradiation tested. This task can commence after two years of sensor development, and will need three years.

Structural Materials

Gap: In Russian LBE technology program, a series of special alloys were developed and tested for enhanced LBE corrosion resistance. The main materials include austenitic EI-211 and ferritic EP-823 with Si addition and special secondary treatment. In recent tests performed by international programs either in Russia or at home institutions, a class of "nuclear" steels, including austenitic SUS 316L, D9, ferritic and martensitic T91 (9Cr-1Mo), HT-9, have shown promising corrosion resistance. However, systematic test over temperature and flow conditions in proposed reactor concepts, especially long-term tests and in-pile tests have not been performed. Effects of radiation on the protective oxide films are not well known.

Some evidence of material lifetime reduction (LCF aging) of ferritic/martensitic steels in LBE (unirradiated) has been observed and will require further evaluation.

Although there is no report of liquid metal embrittlement (LME) in the oxygen controlled regime in Russian LBE technology, there is emerging experimental evidence that LME may occur when oxygen is depleted (such as in abnormal operating conditions and during accidents) or that a hydrogen concentration influence is present.

R&D: Systematic testing of candidate materials for vessel, in-core materials and fuel cladding needs to be performed. Tests can be roughly categorized in two classes: short to medium time interval (up to 3000 hours) tests for initial oxidation and corrosion processes, and long time (over 6000 hours) tests for long-term corrosion behavior. The first class of tests should also include variations of oxygen activity in LBE, the study of the growth and repair mechanisms of oxide films, and the upper flow velocity limit before the onset of erosion. This is a 5-year task, will require testing facilities (e.g., DELTA Loop at LANL) and operating funds. A large test facility has the added benefit of testing some components, such as pumps, flow meters and valves at the same time.

The study of LME can become very involved. Initial study will include tensile testing of candidate materials in a range of oxygen activity, and the wetting characteristics of LBE to steels. This is a three year task.

Aging evaluations vs environmental conditions will also be investigated.

Preconditioning of Structural Materials

Gaps: For materials used in environment at the high end of the reference technology (above 500°C), it is necessary in some cases to precondition them, i.e., pre-oxidize them so that the kinetics is favorable for growth of protective oxide film during operations. There has been little systematic evaluation and development in this area.

R&D: For promising candidate materials, especially the ferritic and martensitic steels for fuel cladding and other high temperature applications, perform preconditioning (e.g., hot dipping in oxygen saturated LBE bath) tests and subsequent corrosion testing in lead-alloys. Characterize the improvement over materials without pre-conditioning.

Extension of the Reference Coolant Technology

The high potential Generation IV reactor concepts typically call for conditions that are different from the loop type compact Russian reactor designs. This is to aim for enhanced performance and improved economy by full utilization of the unique properties of lead-alloys. It introduces uncertainties in the coolant technology and materials.

Special Surface Treatment of Structural Materials

Gaps: Within ADS development, several special surface treatment methods have shown promise of enhanced corrosion resistance. They include procedures to aluminize steel surface via heat treatment and electron beam. Initial testing (both static and dynamic) shows exceptional resistance to LBE. However, it is not known how its performance will change if the film is accidentally damaged, and how it behaves in radiation environments.

R&D: For specially treated materials, tests will be performed in a range of operating and accident conditions, with emphasis on coating stability and response to damages. If it proves to have sufficient lifetime performance, then in-pile testing will be performed. This could last for 5 years.

Oxygen Control in Large Systems and Natural Convection Driven Systems

Gaps: The Russian technology was developed and deployed in loop type compact reactors, and has been tested in flow loops outside Russia. However, many Generation IV lead-alloy cooled reactor concepts call for natural convection and/or hybrid driven coolant system in large open lattice vessels. The flow characteristics, and the oxygen and corrosion products transport are vastly different from that of the proven systems. The effects of local abnormality of coolant chemistry and induced flow instability (e.g., through change of heat transfer properties) are not known.

R&D: Testing is needed to study the efficacy of oxygen control in large systems and natural convection systems. This will require the construction of pool type or modification of loop type test facilities with local flow and coolant chemistry measurement capabilities. This is a 5 year task, and needs $5M level total funding. (A close cooperation with the European countries, where large loop and pool type facilities are in operation or in construction, will be helpful for any non-European countries electing to pursue L6/L4).

Coolant Technology and Materials for Lead Coolant

Gaps: The proven technology (in Russia) is for LBE. In the material testing program for BREST (MINATOM, Russia), the application of the LBE coolant technology to lead resulted inconsistent and decreased performance.

R&D: Run tests of the coolant technology and materials for LBE in lead, starting with static tests, then dynamic tests in flowing lead test loops and large vessels. This is a 5-year task, and needs $5M level funding.

Materials for Long Life Cores

Gaps: Under existing operating experience and loop testing, there is some indication that very long lifetime may be achieved with EP-823 type alloys under the conventional operating conditions. There is, however, no sufficiently long test to establish that, since some Generation IV long life core designs call for 12 to 18 years of service life (up to 100,000 hours or more).

R&D: A dedicated test loop will be needed to test candidate materials for very long periods of time (over 18,000 hours) with constant coolant chemistry. This is a 5-year task, and needs $5M level total funding.

The oxygen control coolant technology may have some intrinsic limit on how high the operating temperature can go since oxidation kinetics is greatly accelerated at high temperatures. To achieve the high performance and versatility of the proposed Generation IV reactor concepts, there are critical needs to re-examine the coolant technology and to screen for new cladding materials.

Screening of High Temperature Materials

Gaps: The highest operating temperature with the current set of structural materials suitable for lead-alloy coolant is under 650°C, much lower than the 800–900°C fuel cladding temperatures envisioned in the high-temperature Generation IV reactor concepts.

Using steels as the main structural materials, the existing LBE technology requires a proper control of the oxygen level to mitigate the steel corrosion problem. Under this framework, if oxygen is depleted, liquid metal corrosion via dissolution attack, and possibly liquid metal embrittlement, can occur. However, at high temperatures in Pb, oxidation kinetics may be accelerated too much and becomes detrimental. Within this higher temperature range, the mechanical properties of some refractory metals and alloys improve but oxidation problems compound (e.g., internal oxidation of Nb). So oxygen-free coolant technology may be needed for high temperature reactors.

Only very preliminary screening has been performed in Pb at high temperature. Experimental results from the 1950s to 1960s were not very useful in this regard because of the poor control and characterization of the coolant chemistry. A broad ranging screening of materials is needed—and especially opening the options beyond alloys alone to include modern materials such as ceramics and composites—taking advantage of work from the fusion community and from the aerospace community. Fabricability and expense are to be considered equally with thermal, chemical and structural issues.

Technology Development for New Fuels

The new fuels under consideration include TRU-Nitride and Th based metal alloys. The choice of clad material and the pin performance requires to develop the coolant/clad/fuel pin as an integrated system. The work will start with materials screening tests for the cladding and with fabrication technology development and thermo/physical/properties testing of the (unirradiated) fuel.

R&D: Start with small scale static tests, with monitoring and control of the oxygen levels in Pb, to screen for high potential candidate cladding and structural materials. Possible candidates include some refractory metals and alloys[g] (Nb, Ta, W), ceramics (silicon nitride and carbide) and composite materials developed in aerospace industry. If promising candidates are identified, small scale dynamic tests in flow loops will be performed going on in parallel.

Once a set of cladding options are selected, they will be evaluated for fabricability. Irradiation test pins will be produced—preferably using a candidate fabrication method.

The lead irradiation tests (likely to be static capsule tests of coolant/fuel/clad combinations at temperature) will help further refine the down selection of leading candidates. This will be about all that can be accomplished during the viability R&D time interval. It will be enough to show that one or several candidate combinations of fuel/clad/bond/coolant appear to have promise.

g. Screening includes neutronics effects, which may exclude some refractory alloys owing to neutron absorption properties.

Subsequent to establishing viability of several candidate coolant/fuel/clad combinations for the several service conditions represented in the L6/L4 missions, a major fuel pin development campaign will be initiated in close integration with a corresponding recycle/refabrication/waste form production technology development campaign. This recycle R&D campaign is discussed in Section 4.4.9.

4.4.2 Neutronics and Control R&D

The neutronics and control function maintains a quasi-static balance of heat deposition and heat removal rates even as fuel burns out over time and as heat delivery demand varies between the startup and full power load schedule of the Balance of Plant. Much of the reactor core design including neutronics, heat removal, thermo/structural feedbacks, control, and overall safety strategy becomes integrated by the design process of satisfying the neutron balance functional need because this design process must actively link fuel, heat removal, heat transport and instrumentation and control issues.

Neutronics Technology Gaps

Technology gaps pertaining to "Maintaining Neutron Balance and Control" are linked to fuel related technology gaps discussed above via fuel and fuel assembly design. The first technology gap pertains to the heterogeneity of the fuel and/or fuel assembly lattice proposed for L6/L4 concepts (e.g., hybrid mixed uranium-plutonium oxide and uranium fuel assemblies, and the so called neutron-streaming fuel assemblies used in L4 concepts). Exact geometrical modelling of such heterogeneous lattices poses challenges to core design methodologies which may require critical assembly experiments but certainly requires continuous Monte Carlo validation of deterministic design codes.

A second example of core design impacts from fuel related innovations is the high content of minor actinides or transuranics in the fuel (going to the extreme of fertile-free fuel). For these fuels, nuclear data either do not exist, or the uncertainties associated with their measurement and evaluation are high. The L6/L4 concepts are designed for small burnup reactivity loss and have low excess reactivity margins; therefore the sensitivity of the neutronics calculation to data related uncertainties is increased. Hence, the core design methodology faces considerable challenges when dealing with these fuels. Finally, given the proposed innovative fuels, and closed fuel cycles with feedback of irradiated fuel, the core design process is faced with uncertainties from lack of knowledge of the beginning-of-life fuel compositions (and additionally, for recycle concepts, after each recycle step) due to gaps of knowledge regarding efficacy of the fabrication, recycle, and re-fabrication technologies. The bottom line is that the challenges for "Maintaining Neutron Balance" stemming from uncertainties linked to fuel technology, basic nuclear data, and recycle technologies are important because these uncertainties get compounded during core burnup—yet must be treated by design neutronics analyses.

Similarly as for the fuel, most structural materials and coolant technology gaps (mostly falling into the category of compatibility between HLMC and structural materials) and basic thermo/physical properties of coolant, clad and fuel have a direct bearing on the neutronics because core design efforts link the heat deposition (neutronics) to thermal hydraulics analyses. There are important methodology gaps for HLMC thermal hydraulics analyses, leading to uncertainties in the results of the respective codes (so called CFD codes). Additionally, some of the innovative design features (e.g., wrapper-less fuel assemblies) will necessitate experimental and theoretical validation of the CFD methods. Here too, "Maintaining Neutron Balance" has to deal with compounding of uncertainties of the base technologies.

The long-life "battery" (or "cartridge") reactor cores of concept set L6 are designed to maintain a constant fissile fuel content via an internal conversion ratio of identically unity. Thus, only a very small excess reactivity margin is to be built in and compensated for at beginning of life. Reactivity changes due to burn-up and power variations are compensated by temperature feedback effects. Some concepts are designed to allow for passive load follow. The challenges such designs pose to the neutronics analyses are manifold. The designs can afford only small uncertainty margins: the temperature distributions must be known with high accuracy (which is not trivial, given the lack of knowledge in the area of HLMC thermal hydraulics), in order to be able to attain a low uncertainty when calculating the temperature reactivity feedbacks. Moreover, high accuracy must be attained in the evaluation of the fuel and HLMC temperatures, as well as the thermo/structural reactivity effects, since L6 concepts often display unconventional reactivity behaviour (e.g., positive HLMC expansion reactivity effect). At the same time, the requirements vis-à-vis the nuclear data are also stringent (lead and lead-bismuth, and, if applicable, minor actinides). The design goal of almost constant fissile fuel content over the core life-time requires high accuracy in the depletion/evolution calculation, which, in turn, poses stringent requirements on the allowable uncertainty margins as regards data and codes. Unconventional reactivity control mechanisms, e.g., reflector movement, or tungsten control rods, require targeted methodology validation efforts and reduction of uncertainties for the tungsten nuclear data.

From a general point of view the L6/L4 concepts strive for reduced design margins and a high degree of optimisation. Hence, comprehensive validation efforts (both theoretical and experimental) are required for the core design methodologies (data and codes), since the reduced design margins do not permit significant extrapolation from the proven domains.

Neutronics R&D Plan

Considering the technology gaps, the scope of an R&D program geared towards L6/L4 concepts should include the following:

- Establishment and validation of evaluated data files (data for thorium/uranium fuel cycles, data for HLMC, data for all the transuranics, their activation and decay products, and fission products to be included in isotopic evolution calculations). This comprises analytical and experimental (XS measurements, critical experiments) R&D efforts.

- Systematic analysis of nuclear data and of heterogeneous lattices and of reflector controlled core designs comparing for deterministic vs. Monte Carlo methodologies, and evaluation of the remaining uncertainties in design methods.

- Analysis of the effect of uncertainties due to nuclear data on operational and safety related core parameters.

- Comprehensive review of the existing experimental basis (critical experiments and fuel irradiation experiments) and establishment of coherent data bases. These data bases will permit to re-analyse, if needed, some of the old experiments, and define new experimental R&D with the overall objective of validating data, codes and methodologies. (It is unclear at this point if fast critical experiments will be necessary during Viability R&D—although there is no doubt of need during performance R&D.)

A few specific issues are summarized here:

At the unit cell level, in deterministic methodologies, the scattering matrices often ignore (n,2n) (and obviously also the higher (n,Xn)) reactions. However, in material compositions with high lead or lead/bismuth content, as well as ^{232}Th, or ^{233}U content, and in the energy range of interest (roughly 1 keV to 5 MeV) for the innovative concepts considered, the (n,2n) processes can be rather important and should be included in the methodology. Another problem without convincing solution in deterministic codes is broad resonance self-shielding. Again, for material compositions with high lead, ^{232}Th, and/or ^{233}U content, this problem increases the uncertainties of the results of the neutronics analyses. Finally, deterministic methodologies will have to apply theoretically well-founded and validated approaches for modelling anisotropic diffusion effects (relevant, e.g., for concepts relying on so-called neutron-streaming fuel assemblies).

At the full core level, the calculation of the evolution of the material compositions is a much more demanding task in the case of the innovative concepts considered here, since, basically, consideration of the full decay chains of minor actinide rich fuel compositions is required. Moreover, all geometry (often heterogeneous) and neutron spectrum effects must be taken into consideration at each step of the evolution calculation, making the whole procedure sensitive to feed-back effects: e.g., uncertainties in the isotopic composition (due to the evolution/depletion methodology (approximations in the decay chains, for instance), or to geometry effects, influence the spectral distribution, which, in turn, will influence the result of the evolution calculation.

Obviously, the availability and quality of the nuclear data (evaluated data files) for previously seldom used isotopes, both for fuel and coolant compositions, is a big issue for the neutronics analyses of the concepts under consideration.

Control Technology Gaps

The L4 concepts are targeted to burn high actinide (plutonium and minor actinides) content fuels and/or “dedicated transmutation fuels” (i.e., uranium-free fuel in which the actinides are embedded in an inert matrix). Fuel without any fertile component are characterized by a very low delayed neutron fraction (β_{eff} value) “, as well as by a very low (negative) Doppler reactivity effect. These neutronics characteristics pose serious challenges to reactor control. That is the reason why Th was introduced into the fuel choices for the L4 concepts. The technology gaps affecting reactor control are thus a consequence of the details of fuel composition choices made for these concepts, and are related as well to neutronics issues (e.g., basic data).

The L6 Concepts employ a passive load follow design strategy to make them practically immune to compounding errors in the BOP and/or control room, as well as to achieve simplification and cost reduction. This overall control design strategy is closely linked to design of the thermostructural feedbacks and to the overarching safety strategy. It will require conceptual work tightly linked to all facets of core design.

Control R&D Plan

The scope of R&D activities as regards reactor control has to rely on all the activities defined in the area of “Maintaining Neutron Balance”. Additionally, when considering L4 concepts, the scope of an R&D program should include:

- Establishment and validation of the evaluated data files, with particular emphasis on kinetic parameters, for all the transuranics, their activation and decay products, as well as the fission products. This comprises analytical and experimental (XS measurements, critical experiments) R&D efforts.
- Numerical and experimental validation of kinetics and dynamics codes.

Passive Load Following

R&D will be directed to establishing passive load follow for L6 concepts. Here the zero burnup reactivity loss eliminates need for an active control rod; if natural circulation is employed, an active pump speed controller is eliminated; then the power is made to self adjust on the basis of the cold leg temperature and mass flow rate of the heat transport loop to the BOP. At the same time, the thermostructural reactivity feedbacks must be such as to keep the reactor temperature in a safe range over the full physically achievable range of these two independent variables – whether they are planned or spurious.

This passive load follow development must be performed as an integral part of the reactor design and safety strategy development. It will be conducted by design analysis during the Viability R&D phase—as the needed basic properties data become available from other areas. The thermostructural design of the lattice assemblies and their clamping in the core structures will be optimized to produce the required thermo/structural reactivity feedbacks. The goal for the completion of Viability R&D will be to have displayed (by dynamic simulation) that this design strategy can be achieved and that it is robust with respect to anticipated evolutions of properties as the fuel burns and the plant ages. It will be required to confirm this by tests in the prototype during later stages of the Gen IV development campaign.

4.4.3 Lattice Heat Removal R&D

Technology Gaps

The L4 concepts operate at reduced power density compared to traditional Na-cooled fast reactors; the L6 concepts operate at significantly reduced power density. In both cases a more open lattice is employed so as to reduce pressure drop across the core and reduce associated pumping requirements; (in the case of the L6 concepts, natural circulation is used). In all cases the Pb or Pb-Bi velocity is maintained at under 2 m/sec to avoid coolant erosion of protective layers on the surfaces of the fuel cladding and structures.

These conditions give rise to many new issues of lattice heat removal. First, the open lattice precludes the use of pin wire wrap for pin spacing—a support grid is required as in LWR’s. The structural integrity of this grid over long life is a development issue. Moreover, were the coolant chemistry to be allowed to drift out of the control range, a possibility for crud buildup on the grid (and on other locations in the flow path) cannot be ruled out, and the avoidance of hot spots must be demonstrated. In this regard a ductless assembly design is contemplated.

None of these lattice heat removal issues is a potential showstopper, and all will be resolvable by a structured multi-year experimental program using Pb or Pb-Bi flow loops and an associated program for development and validation of appropriate 3D, transient computational fluid dynamics computer codes.

Lattice Heat Removal R&D Plan

Computational design tools for open lattice, ductless assemblies with grid spacers must be developed and validated in order to deal with this geometry. Such validation will require the existence of an extensive set of experimentally determined pin bundle heat transfer correlations and pin bundle/grid spacer pressure drop correlations as functions of geometry, pin linear heat rate, and flow rate. Such experimentally derived correlations of adequate scope do not currently exist and must be generated.

Oxide film layers on the pin surface—used to control corrosion and mass transport—affect the value of impedance to heat transfer from the fuel to the coolant. Dependencies of this impedance on the flow regime and coolant chemistry program must be developed experimentally.

Most concepts rely on natural circulation for removing the decay heat. Proving natural circulation in HLMC loops is an obvious technology gap to be dealt with in the coolant R&D scope. Substantiating the minimum natural circulation characteristics requirements of the concept requires comprehensive transient analyses with implications not only on heat removal but with direct bearing on safety strategy gaps and associated R&D.

4.4.4 Heat Transport and Component R&D

Technology Gaps

Nearly every L4/L6 concept uses a pool plant layout (rather than loop); some employ primary mechanical pumps while others employ natural circulation and still others employ a gas lift pump. Invessel heat exchangers span a range from IHX's with low pressure liquid metal on both sides of the tubes to high pressure invessel steam generators (change of phase and of heat transfer coefficient) to very high pressure supercritical steam or supercritical CO_2 invessel IHX's (no change of phase).

The structural design of the invessel IHX's or SG's will be required to deal with new issues of compatibility with the new fluids encountered; strength at high temperature; fabrication of the new materials which may be required; support given that when the structures tend to float in the coolant; heat transfer and pressure drop correlations for the new fluids; and inspectability and reparability.

Under low velocity flow conditions (especially in natural circulation designs) it will be necessary to take care to avoid stagnant zones where crud would tend to accumulate.

In the cases where the reactor drives a chemical plant, the heat exchangers may experience combinations of especially hostile temperature and corrosive fluids as for example H_2SO_4 at 900°C or HBr at 750°C. IHX material choices and designs for these conditions will provide difficult challenges.

Piping to the balance of plant may need special levels of flexibility to accommodate not only thermal expansion but also accommodate seismic isolation of the reactor vessel. (Three dimensional seismic isolation may be used in some designs.)

Mechanical pumps for Pb-Bi have been in use for Russian submarine service for many years; special impeller designs to deal with erosion have been employed. For Gen IV concepts scale-up and/or higher temperature service and/or longer service life may require further development.

Especially for very high temperature service (800°C) contemplated for some L6 concepts a gas lift pump approach may hold advantage because the blower can be located on the vessel head and can operate at lower temperature. The thermal hydraulics of maintaining bubbly flow in the riser will require R&D as will technologies for phase separation prior to cover gas return to the blower.

The use of natural circulation in a pool plant layout as is proposed for numerous L6 concepts will face the issues of flow stagnation and of thermal stratification—especially in the cold pool and especially during power coastdowns. It will be required to do 3D scale model testing to properly design baffles and entrance and exit horns around the heat exchangers to minimize the opportunities for such flow stagnation and thermal stratification.

During power startup under natural circulation a tendency will exist to suffer reverse flow through the lower power outer fuel assemblies—bypassing the overall heat transport loop and heat removal to the balance of plant. The same tendencies for core recirculation flow will exist during transition from full power to low power in load follow transients. Careful design strategies to tailor full loop vs core recirculation pressure drops under a wide range of power and flow conditions will be required in the design. Scale testing will be necessary. Full loop, 3D, transient computational fluid dynamics codes validated against such tests should be developed to assist the design effort.

No issue here is a potential showstopper with the possible exception of finding a proper material which is strong enough, and fabricable enough for the very high temperature (800°C) service conditions in Pb which are proposed for some L6 concepts. None-the-less, a quite substantial development program will be needed to develop and prototype the heat transport components for the new service conditions.

R&D Plan

Heat transport components R&D during the Viability R&D phase should be focused on the highest potential payoff enabling technologies but should also initiate scope work on longer term or incremental cost reduction innovations—which will receive increased focus at a later date. The TWG believes the highest potential payoffs pertain to:

- Heat exchanges for Na and Pb-Bi to Supercritical CO_2 at ~ 550°C
- Recuperators for the supercritical CO_2 Brayton cycle
- Invessel steam generators for Pb-Bi service at 550°C
- Lift pump technologies—bubbly flow regime geometry constraints; gas/liquid separators; and moderately high temperature blowers
- Technologies for piping accommodation of displacements experienced in seismic isolator applications
- Development and validation of 3D transient CDF codes capable of treating natural circulation flows in pool plant layouts.

The development activities are targeted to reach a level of establishing technical feasibility by the end of Viability R&D. It will include materials selections, conceptual designs, and small scale testing such as single tube heat exchanger or steam generator tests, scale water flow tests on lift pump flow regimes, small scale phase separators, etc.

4.4.5 Reactor Structures, Shielding and Refueling R&D

Technology Gaps

The physical, chemical and thermodynamical properties of the heavy liquid metal coolant employed in the lead- and lead/bismuth-cooled reactor concepts result in numerous prominent structural design features placing special challenges before the structural designers of Generation IV nuclear reactor facilities.

First, the coolant's high density places several demands in designing for protection against the effects of seismic events. In general, support and stabilization of coolant-containing pipes and vessels become more complex and costly as their wall thickness and weight requirements increase. Relevant weight requirements might therefore limit the reactor's size for free hanging vessels. In the case of use of secondary loops with lead alloys, it may not be possible to scale up the size of heavy metal cooled reactors due to the difficulties associated with supporting lead-containing vessels and providing adequate flexibility to the piping systems. In the case of the pool plant layout, the use of horizontal seismic isolation support seems to be an effective solution to solve this issue, in fact horizontal isolation can reduce the horizontal seismic loads more than one decade. Further it is known that vertical loads are in general well tolerated by the components of a pool-type reactor owing to their specific cylindrical configuration. An additional issue comes from the sloshing effect of the lead alloy's free level on the internal mechanical structures; in fact, in spite of the strong acceleration reduction associated to the possible use of the isolation supports, the frequency of the seismic excitation can approach the natural frequencies of the sloshing phenomena with residual relevant loads. The development of new models for the interaction between fluid and structures becomes an additional design need. Alternately if monolithic concrete vessels are used, high temperature concrete will be required, or design provision will be required to maintain low concrete temperature in normal and off-normal conditions.

Second, because heavy metal coolants are denser than the structural components that make up the core, the reactor designer must consider special approaches for anchoring parts, and to prevent the fuel, blanket and shutdown assemblies from floating up in the coolant, particularly during refueling. However, the buoyancy of the coolant balances the gravity pull of the submerged portions of major components and vessels thereby reducing their stress level. Since the core has a similar density to lead, it is subjected to relative low loads in the case of a seismic event.

Third, the coolant's high density and opacity demands the development of appropriate in-service inspection techniques and repair capabilities.

Fourth, the corrosion characteristics of molten lead and the high operating temperature range required to prevent the coolant from freezing both contribute to the need for high-performance steels in the primary heat transport system. Utilization of such high performance materials generally require more costly fabrication techniques compared to conventional steels, in addition to their higher commodity cost. Nonetheless, to capture the unique high temperature capability of lead- and lead/bismuth-cooled systems, well beyond what can be achieved in sodium-cooled reactors, is essential if these reactors will play a major role in the nuclear power industry's future.

Fifth, the production of highly volatile, toxic and alpha-active polonium-210 in lead/bismuth coolant represents hazard to the individual and collective physical protection of the operation personnel. Whereas for Na systems strict control of system leak-tightness is necessitated by the chemical reactivity of Na, here the same rigor is imposed by the Po aerosol toxicity.

The L6 systems are designed for transportability which constrains weight, making the design of the shield to ensure long core lifetimes and reuse, for the battery type lead-alloy-cooled reactors a major design concern.

Major potential payoffs, technology gaps and showstoppers with regard to *reactor structures, shielding, and refueling* are summarized below:

Potential Payoffs

Simpler refueling approaches

Elimination of the intermediate heat exchanger

Less stress for internals (less material and reduced cost) resulting from both buoyancy and seismic isolation.

Technology Gaps

Structural steels compatibility with lead and lead/bismuth at high temperature

Material descriptive equations for creep and creep-fatigue damage

In-service inspection techniques and repair capabilities

High temperature design codes for new structural materials

Seismic isolation for siting anywhere.

Potential Showstoppers

Selection of high temperature material vessel support approaches as heat rating and vessel size increase

Coolant and steel compatibility (corrosion).

R&D Plan

Specific R&D needs are given below, where the major potential advantages (payoffs), major potential disadvantages (showstoppers) and the technology gaps with regard to the design of the reactor structures, core shield and refueling mechanisms for the Gen IV reactor systems are identified.

The scope of an R&D program geared towards lead-alloy-cooled reactor structures should focus primarily on high temperature structural design, seismic design and seismic isolation, and security hazards. These issues are discussed next.

High Temperature Structural Design

Lead and lead/bismuth cooled Gen IV reactors will operate at high enough temperatures so that, according to the ASME code, the structural designer must take into account elevated temperature issues. The following issues are of concern from an elevated temperature design standpoint:

- Elevated structural design of reactor internals and coolant module
- Thermal ratcheting of the reactor module and coolant module
- Creep-fatigue design of the reactor module in the region of the hot and cold cross flow
- Shielding design to ensure core lifetime of 15 years and reuse
- Material compatibility with lead
- Leak before break design of coolant module
- Integrity of concrete support with high temperature near the reactor exterior cooling system
- Design of the Feedwater-SG coaxial pipe
- Fatigue-creep-corrosion monitoring system
- Dissimilar metal welding design.

Seismic Design and Seismic Isolation

Since Gen IV nuclear reactors are intended to be deployed worldwide, some plants may eventually be located in seismically active regions. Structural engineers are thus challenged to design power plants capable not only to survive seismic events but also to continue to provide power during and after earthquakes. Passive seismic isolation is the leading candidate for achieving this goal.

The issues relevant to seismic design and seismic isolation are the following:

- Two dimensional versus three-dimensional seismic isolation system design
- Support structure design
    - Reactor support
    - Coolant module, reactor module and containment vessel

- Core support structure

- Core seismic design to reduce core compaction
- Shutdown rod insertion
- Reactor upper closure head design
- Piping design to minimize secondary coolant and BOP working fluid
- Piping support
- Buckling design
    - Coolant module
    - Reactor module
    - Containment vessel.

Security Hazards

As well as functioning as the ultimate (third) line of defense against radioactivity release, the containment building of a NPP protects the reactor and its support systems from external natural events. Additionally, institutional measures such as personnel screening, limited site access, remote monitoring, security forces, pre-planned emergency response actions, etc, are in place at nuclear installations. Should the evolution of threat profiles require enhanced protective measures in the future, these traditional means already provide the framework for any required incremental strengthening of safeguards. Additionally, design for increased operating margins and for passive safety response, both of which are elements of Gen IV design strategy, also increase innate robustness with respect to upset events—both unintended or (in the case of terrorist acts) intended.

4.4.6 Overall Safety Strategy R&D

A hallmark of the L6/L4 concepts is the extensive use of passive safety design approaches. These passive safety approaches are argued to both remove all safety function requirements on the balance of plant and allow for a non-traditional containment building. The approaches have already received substantial R&D effort for the past fifteen years in connection with Na-cooled fast reactor design—especially for concept set L2. They received US-NRC prelicensing review of the SAFR & PRISM concepts which employed them. The additional R&D required in many cases will be of a confirmatory nature as opposed to a developmental type.

Passive Decay Heat Removal

All but one L6 and L4 concepts employ a pool layout with a guard vessel with all primary coolant confined in the ambient pressure vessel with top entry penetrations; the loss of coolant hazard is eliminated by design. Decay heat removal is by natural circulation from the core to the normal heat transport circuit with passive backup via heat removal across the vessel wall, vessel-guard vessel space, guard vessel wall to either a natural draft ambient air channel or to a natural circulation water pipe network in a concrete monolith guard vessel.

R&D needs are confined to evaluations of emissivity vs vessel surface aging; optimization of boundary layer trip surface modifications in the natural draft chimney; wind effects on natural draft chimney effectiveness and chimney interactions; and smooth transition to natural circulation cooling of all core regions under various initial conditions with minimal thermal stratification.

Passive Self Regulation of Reactivity

All L6/L4 concepts employ innate reactivity feedback coefficients to passively maintain heat production in balance with heat removal. This design strategy has been demonstrated in tests at EBR-II, FFTF, and Rapsodie test reactors.

The R&D required here will be directed primarily at the innate radial core expansion (axial neutron leakage) reactivity feedback effect. The ductless assembly, open lattice, and grid spacer pin separators are quite different from the ducted assembly, tight lattice, wire wrap spacer core designs of Na cooled reactors for which the experience base exists. Tests on flow distributions and temperatures in the ductless assemblies will be required. However, thermostructural reactivity feedbacks in response to shifting temperature fields cannot realistically be confirmed until a prototype reactor is built. Additionally Doppler affect of higher actinides must be measured in zero power critical facility neutronics tests.

Burnup Reactivity Swing/Rod Runout Worths

All L6/L4 concepts design for zero or near zero burnup reactivity swing so as to maintain a small rod bank worth and a single rod worth less than the delayed neutron reactivity worth.

The R&D requirements here relate to the currently poorly known basic nuclear data for the higher actinides. Differential and integral experiments will be needed to determine fission and capture cross sections and ν values for the minor actinides as well as their branching ratios and radioactive decay chains and time constants. Until these data are better known, burnup reactivity swing cannot be accurately predicted. Additionally measurements of thermal and fission gas driven axial creep of the fuel must be made.

Coolant Density and Void Reactivity Worths

Coolant boiling in Pb or Pb-Bi cooled reactors does not occur until the coolant reaches ~1700°C—well after structural loss of integrity has already occurred. However, other pathways to local voiding of the lattice can be postulated such as fission gas release through a clad rupture; nitride fuel dissociation; steam entrainment and transport to the core upon steam generator tube rupture; and perhaps others. While the coolant void coefficient of reactivity is generally less positive in Pb or Pb-Bi cooled reactors than in Na ones, it is none-the-less locally positive, so its value is relevant to some postulated accident sequences.

Neutronics critical experiment measurements of local void coefficients of reactivity must be a part of the R&D program. The coolant density coefficient of reactivity must also be determined—as it is a contributor to the overall power coefficient of reactivity upon which passive power self regulation depends.

Fuel/Clad/Coolant Phenomenology Under Severe Accident Conditions

Fuel/clad/coolant phenomenology is a crucial determinant of performance in normal, upset, and severe accident safety considerations. First is fuel/cladding compatibility under normal and upset conditions; mechanical or thermo/chemical interaction which could fail clad integrity are to be avoided. Next is the coolant/clad phenomenology. The clad integrity should not be challenged by coolant chemical or mass transport degradation mechanisms. Next is fuel/coolant compatibility; in the event of a clad breach, the fuel coolant interaction should not form low density products which block off coolant flow, and fuel dissolution or dispersal in the coolant is undesirable—satisfying both requirements allows for safe run beyond cladding breach to end of normal reload cycle. Lastly the fuel/clad/coolant phenomenology during a severe overpower or undercooling event should not exacerbate the event or lead to autocatalytic effects.

Much of the safety-relevant R&D needed here has been discussed already in previous sections. Here it is sufficient simply to enumerate the known issues. First, coolant chemistry control is critical to avoid coolant chemical attack on the clad—forming crud that could collect as sludge deposits at coolant channel entrances or at grid spacers and choke off flow.

Next, fuel/coolant interactions under run beyond cladding breach will require testing. While nitride fuel is known to be compatible with Pb or Pb-Bi, uranium containing metal alloy is known to be slightly soluble; degree of solubility and place of plateout will require testing.

Upon fuel pin disruption, all fuel and cladding are expected to float in the coolant and vapor explosions are highly unlikely in light of the extraordinarily high coolant boiling temperature. However the dynamics during the "transition phase" (after clad breach but before final dispersal) are complex—involving pressure driven dispersal both upward and downward, inertial impedance of dispersal due to the high specific gravity of the coolant, solidification of clad and/or fuel on colder axial segments of the fuel assembly, radial propagation of disruption, flow blockage, flow redistribution, reverse flow, etc. For nitride fuel a potential for dissociation and gas-driven pressurization and coolant acceleration and impact on vessel structures has been postulated.

To fully characterize the severe accident fuel clad/coolant phenomenology will demand an extensive inpile transient testing R&D campaign and will be essential to resolve licensing issues.

Steam Generator or Intermediate Heat Exchanger Tube Rupture

The relative (compared to Na) chemical inertness of Pb or Pb-Bi coolant in contact with air and water has been exploited in some L6/L4 concepts by eliminating the intermediate heat transport loop. In some cases the steam generators are placed in the primary vessel (and in some cases, the steam pressure is supercritical). In other cases a supercritical CO_2 heat exchanger is placed in vessel to drive a Brayton cycle. In some cases a low pressure heat exchanger is placed invessel to drive a process heat or a water cracking chemical plant. The high pressure fluid inside the low pressure primary vessel and separated from it by a single wall heat exchanger tube presents a hazard for which no experience base carries over from Na cooled fast reactor experience.

Entrained flow test rigs at significant scale will be needed to evaluate measures taken to assure that no gas voiding of the core lattice can result from invessel steam generator or heat exchanger tube rupture.

Pb Coolant Solidification

While the Pb-Bi melting temperature at 123°C is near that of Na (93°C) for which a substantial experience base exists, for concepts using Pb, the coolant freezing temperature at 327°C is rather close to the Rankine cycle feedwater temperature. (The required ~100–150°C coolant temperature rise needed to keep steam generator (SG) and invessel volume acceptably small and the upper bound cladding temperature of ~550°C for current materials leaves little room between coolant inlet and freezing temperatures.) Accident scenarios such as feedwater heater failure raise a hazard of coolant slush/stringer formation with a potential to block flow channels. Not all concepts experience this issue because they are either Pb-Bi cooled or (e.g., M17) don't suffer from this freezing pinch point owing to a much higher outlet temperature and a much higher He intermediate loop cold leg temperature.

Large scale testing will be needed to assure that the ductless, open lattice of L6/L4 designs will assure adequate cooling even in the face of slush formation—or that design measures provide for suitable trace heating to preclude freezing even under upset conditions.

R&D Plan

The safety strategy, which incorporates passive safety response, requires tight integration with reactor design at every design decision point. As a result, and as enumerated in the listing of safety – relevant technology gaps, a majority of the R&D needed to support development of an overarching safety strategy are performed for the numerous other functions already discussed earlier.

Several areas are specific to safety, however; those often pertain to understanding phenomenology under beyond design basis (severe) accident conditions.

- Nitride fuel volatility phenomena under hypothetical high temperature conditions should be understood first on a unirradiated materials basis, then on as fabricated basis at several levels of burnup
- Crud blockage phenomenology on core inlet structures and grid spacers as a function of coolant chemistry, flow velocity, and temperature must be studied experimentally
- Integral neutronics critical experiments using prototypical compositions and geometries will be needed to reduce uncertainties in a host of safety-related parameters:
  - Critical mass
  - Reactivity coefficients
  - Void worth
  - Delayed neutron fraction.

This is especially true for thorium-based fuels. The experiments will be linked to the validation of neutronics modeling and design code veracity.

In cases where very high pressure BOP working fluid is introduced into in-vessel heat exchangers, work is needed on the phenomenology of a tube rupture and on mitigation measures to prevent vessel over pressurization and bubble carryover into the reactor core. This will require both modeling developments and testing.

4.4.7 O&M Strategy R&D

Technology Gaps

The L6/L4 concepts rely on passive reactivity feedbacks for safety functions and the L6 concepts rely on them additionally for passive load follow control. Just as with active systems, it is necessary to periodically "calibrate" and confirm that the values of the feedbacks lie within the design range—even as the fuel burnup level changes and the plant ages. Methods to infer feedback values by measured power response to external perturbations must be developed (using computational simulations of the plant)[h] during viability R&D.

h. The procedures can be demonstrated on the later prototype.

Invessel inspection of structural components is necessary also to confirm the integrity of the core support structures which guarantee the thermostructural feedbacks will function and which guarantee the vessel and guard vessel maintain their leak tightness. Common issues are shared with the L1/L2 Na systems (both Na and Pb are opaque; both primary coolants contain radioactive isotopes), but the density difference may differentiate the ISI technologies.

As a means to reduce staffing at the plant site the L6 concepts rely on a regime of remote monitoring by specialist maintenance teams deployable from the regional fuel cycle center servicing dozens to hundreds of distributed power plants. Such remote monitoring technologies are not uncommon in other industries but will have to be adapted to nuclear service conditions. In the case that some monitoring may be part of the nonproliferation regime, technologies for assuring integrity of the communication link will be needed.

None of these gaps is a potential showstopper.

R&D Plan

The work to develop a testing regime to confirm the values of safety-related reactivity feedback was worked out during the IFR program and was presented to the NRC during their Safety Evaluation Review of PRISM (ALMR). These plans should be adapted to the L6 situation where passive safety is extended to encompass passive load follow as well.

The work on ISI will be cross cutting with similar development for L1/L2—but adapted to the case of high specific gravity coolant and (for some concepts) higher temperature.

Remote monitoring and secure remote monitoring technologies need only be adapted from existing technologies.

4.4.8 Fabricability and Capital Cost Reduction R&D

Technology Gaps

The L6/L4 concept sets hold the potential for capital cost reductions via a wide range of strategies. The more obvious ones include:

- Elimination of the intermediate heat transport loop owing to the relative chemical compatibility of Pb-Bi or Pb with water.
- The use of modularization/factory fabrication for small or mid sized plants.
- The use of passive safety to reduce the number of safety grade systems and especially to remove all safety functions from the Balance of Plant—allowing it to be built and maintained to industrial rather than nuclear safety standards.

Additional high potential payoff strategies introduced in the various L6/L4 concepts include:

- Abandoning the Rankine steam cycle in favor of the simpler (less equipment) and more efficient supercritical CO_2 Brayton cycle.
- Broadening the energy product mix to include cogeneration products and storable energy products—which in some cases move the capital cost of the energy converter to the end user (e.g., manufacture hydrogen and move the capital cost of the fuel cell to the client).
- Replacing mechanical pumps with simple gas lift pumps.
- Using direct contact heat exchange or close coupled intermediate heat transport loops.

Finally, all L6 concepts are based on an approach of serial factory fabrication of small transportable turnkey plants of short on-site construction period with full fuel cycle service provided from a regional fuel cycle center—removing the need for investments in indigenous infrastructure.

All of these strategies for capital cost reduction will require development and frequent reassessment of their potential for economic payoff. Most of them are not concept specific even within the L6/L4 concept sets; they are strategies with broad crosscutting applicability to numerous Gen IV concepts.

R&D Plans

One of the innovations of highest potential payoff to all liquid metal concepts is the supercritical (SC) CO_2 Brayton Cycle which even at only 550ºC core outlet temperature achieve ~45% conversion efficiency and which dramatically simplifies the BOP and reduces capital cost. It should receive high priority development in the Gen IV Viability R&D program:

- Completion of thermodynamic evaluations
- Equipment and piping material selections
- Design and small scale testing of Pb or Pb-Bi to SC-CO_2 and of Na to SC-CO_2 intermediate heat exchangers
- Design and small scale testing of a SC-CO_2 recuperator and turbine
- Design of a 150-MWe prototype SC-CO_2 Brayton cycle BOP.

Another of the innovations of highest potential payoff to numerous high temperature Gen IV concepts are the several thermochemical water cracking cycles for the manufacture of hydrogen. The Ca-Br (modified UT-3) cycle is targeted for use with L6/L4 concepts—operating at 750 to 800°C core outlet temperature. The chemical process itself should receive high priority development in the Gen IV Viability R&D program:

- Identify materials for IHX service at 800ºC for heat transport from Pb to an intermediate fluid (either He or CO_2) and from an intermediate fluid to a reactant mix of HBr and steam. The search for materials should include not only refractory alloys but also ceramics and fiber composites and the downselection criterion should include fabricability and cost as primary considerations.
- Perform the chemical engineering research needed to determine:
  - Rate constants
  - Durability of supports
  - Thermo/chemical/physical properties of reactants and their mixtures
  - Separations efficiencies
- Conduct bench scale demonstration and optimization of the process
- Design a prototype plant at the 400 MWth level
- Continually interface with the nuclear plant design team to address the coupling and the safety issues of a co-sited nuclear/chemical plant.

The R&D plans for energy conversion are described in Section 4.4.10.

Unless it becomes possible to achieve economy of serial factory fabrication of small transportable turnkey nuclear heat supply plants, the L6 concept set is a nonstarter. While not a research topic, this is certainly a crucial development area during the Gen IV Viability R&D campaign. It will be necessary to create opportunities for nuclear plant designers and fabricators to learn and apply what has been accomplished in other industries which base their economic case on serial factory fabrication of large, complex structures and components. This is a challenge for industrial cross fertilization and institutional innovation which might be accomplished by steps such as:

- Multidiscipline expert workshops
- University Industrial Engineering Department special courses provided for nuclear engineering students
- Professional Society Plenaries and Special Session Themes and Workshops
- Participation in Gen IV of foreign industries, which are already ahead of the U.S. on these technologies (e.g., Toshiba, Mitsubishi).

4.4.9 Fuel Cycle [Recycle, Refabrication, and Waste Forms] R&D

Technology Gaps

All of the L6/L4 concepts are fast neutron spectrum multi recycle systems intended to achieve essentially 100% fission of the actinide feedstock and to send only fission products (and trace losses of actinides) to the waste repository. Given that they all recycle a mix of plutonium and minor actinides, they all employ remote fabrication of intensely radioactive fuel feedstock.

The combination of fuel, coolant, and service (temperature) conditions in the L6/L4 concept sets is displayed in Table 13.

The fuel types are Th/U/Pu/MA/Zr or Th/MA/Zr alloy in L4 concepts; and TRU nitride or U/TRU/Zr alloy in L6.

The proposed recycle methods are advanced aqueous, pyroprocess, or another dry process.

The refabrication is based on vibrocompaction, simplified pellet fabrication, or injection casting.

The coolant/clad/fuel combinations are all new ones. Therefore, in all cases, the recycle and refabrication technologies require substantial additional development and prototype demonstration prior to commercial deployment.

Finally, in all cases a fuels irradiation and safety testing campaign must accompany the recycle/refab development. In every case (even the oxide and metal) fuel forms containing significant minor actinide components have not received extensive steady state and transient irradiation testing up to now.

The R&D required for the recycle/refab and irradiation testing of oxide and U/TRU/Zr alloy fuels is treated in detail in the L1/L2 sections of this R&D Scope Report. Here we will focus on R&D required for the nitride and the Th bearing Zr alloy fuels, and it will be treated as incremental R&D—piggy backing on the work for L1/L2 and utilizing many of the same facilities.

Nitride Fuels: Technology Gaps

Nitride fuel has been chosen for service with Pb or Pb-Bi coolant because of its chemical compatibility with the coolant and because of its favorable high density (helping to achieve high internal conversion) and high thermal conductivity (helping to achieve passive safety response). Nitride fuel has exhibited low fission gas release with burnup. Enrichment in nitrogen 15 appears to be necessary for two reasons, first to reduce negative reactivity contributions from neutron absorption on ^{14}N and second to avoid generation of radioactive carbon 14, a gaseous activation product.

Nitride fuels for fast neutron reactor application were studied briefly in the 1970's. They were selected for use in the lithium cooled SP-100 space power reactor program at very high temperature service conditions, and a U nitride properties testing and irradiation testing program was initiated. Irradiation testing of nitride has not exceeded discharge burnups of 4 to 5 atom percent.

The irradiation performance database with (U,Pu)N is sparse, so initial irradiation testing will investigate potential limits of current designs and identify design improvements. The performance of nitride fuel will be coupled to identification and development of new cladding alloys or composites for high-temperature 800°C strength and swelling resistance in Pb service.

Nitride fuel cycle technology is currently not well established, although work performed to date (mostly in the 1960s through the 1980s) indicates promising prospects. Nitride can be reprocessed by both aqueous and dry methods and both are under active development in Japan and in Russia. For example in the double strata concept foreseen in the OMEGA program, JAERI investigates the possibility of using nitride fuels and pyrochemical process in the second stratum. Since the mid-1990s, JAERI has been working on a fuel cycle combining nitride fuel with pyrochemical recycling methods in molten chloride media at a laboratory scale. Although encouraging preliminary results have been obtained with uranium, major R&D work remains to be done on the transuranium nuclides.

Table 13. Coolant/clad/fuel combination and recycle/refab choices concept set.

Coolant Outlet Temperature	L4		L5		L6	
~550°C	Coolant	Fuel	Coolant	Fuel	Coolant	Fuel
	Pb-Bi	Th/U/TRU/Zr alloy TH/MA/Z alloy	Pb	Nitride TRU Nitride TRU Oxide	Pb-Bi	U TRU Zr alloy TRU Nitride
	Recycle	Refab	Recycle	Refab	Recycle	Refab
	• pyro	• injection casting	Dry [TBD]	Vibro	Pyro or Adv. Aqueous or dry [TBD]	Injection casting Vibro (Nitride)
~800°C					Pb	TRU-Nitride
					Recycle	Refab
					Adv. Aqueous or pyro	Vibro Vibro

Much of the work to be done includes adaptation of developing technology for oxide aqueous or dry remote fuel fabrication with increased minor actinide contents to nitride.

Table 14 summarizes the state of technology for nitride fuel. To summarize:

- The fabrication experience with vibropacking is limited and needs further development.
- A low-cost technology to enrich the ^{15}N component is needed to improve the economics of the fuel cycle.
- Irradiation testing is quite limited and often not well documented.
- Phenomenological characteristics affecting basic fuel design such as swelling, fission gas release, fuel-cladding chemical interaction, and thermal dissociation are not well known at higher burnups.

Table 14. Nitride fuel development.

	Items	A. Status of technology	B. Status	C. R&D issues
(1)	Fuel specification	[Fuel type (fabrication experience)] • Pellet/helium bond (over 1,000) • Pellet/liquid metal bond (tens) • Low "smeared density" fuel is understood as typical high burn-up fuel concept to have benign FCMI, but no firm specification.	• Vibropacking fuel, (TRU fuel with FP) • Vibropacking fuel, (TRU fuel) • (Pu enrichment: 8wt.%, TRU fuel)	[High burn-up fuel specification] • Same as for (3) below
(2)	Nitrogen isotopic composition	• Requirement of N15 enrichment to exclude N14, the source of C14 by [N14(n,p)C14] • Conventional technologies of N15 enrichment - NITROX - Ion exchange, etc.	• It is desirable to use enriched N15 for nitride fuel. [M16]	[N15 enrichment technology] 4 Economical enrichment technology development - Ex:Zeolite PSA method
(3)	Burn-up	[Irradiation experience] • Sodium bonded fuel : ~160GWd/t (at peak) • Helium bonded fuel: ~70GWd/t (at peak) • High burn-up fuel concept is indispensable to realize economical advantage. • Irradiation experiments indicated significant FCMI at extended burn-up due to high fuel swelling rate and low creep rate.	• Average discharge burn-up: □6at.% (UO_2 fuel core) • Average discharge burn-up: over 10at.% • Maximum burn-up : 105GWd/tHM (Metal fuel core) • Average discharge burn-up: 72MWd/kg, Maximum burn-up: 121MWd/kg • Maximum burn-up: 15-20at.% (Metal fuel core) (This value is based on the irradiation achievements of metal fuel.) • Average discharge burn-up: 100MWd/kgHM (Metal fuel core)	[High burn-up fuel specification] • High burn-up irradiation test of various type of fuel pin to identify the high burn-up fuel concept with benign FCMI Examples: Large gap with LM bonding Annular fuel with gas bonding etc. • [Core material develop-ment] • Core material development for high neutron dose to achieve high burn-up Ex.: High strength ferritic steels including ODS (oxide dispersion strengthened ferritic)
(4)	Core safety	Possible candidate approach 1) Utilization of inherent and/or passive shutdown mechanisms 2) CDA mitigation (Nitride fuel high temperature dissociation behavior is one of the critical issues.)	• This reactor has high passive safety characteristics, that limit the fuel temperature in accidents. (No description of thermal dissociation behavior) • This reactor has high passive safety characteristic, that limits the fuel temperature in accidents. Some concerns of nitride fuel high temperature dissociation behavior.	[Inherent/Passive safety] • (Precise investigation is indispensable to assure the feasibility.) [CDA mitigation] • In-pile or out-of-pile high temperature dissociation test of nitride fuel • Transient test to evaluate fuel pin failure limit

A considerable amount of research and development will be required to bring the status of nitride fuel up to that of either metal or mixed oxide fuel. But nitride fuel does appear to have unique safety characteristics that are superior to that of either mixed oxide fuel or metal fuel [2]. And for selected high temperature applications such as space power, nitride fuel has unique advantages, and irradiation testing results have been positive.

Nitride Fuel R&D Plan

The clad compatibility and irradiation testing components of this plan were addressed in broad scope in Section 4.4.1. Here we discuss the tight links between fuel pin performance and fuel fabrication (controlling morphology), which is in turn tightly linked to recycle technology (controlling composition).

Recycle Technology R&D

Active programs already exist in Russia (for the BREST concept—in L5) and for the ADS (Omega Program) in Japan. TRU/Nitride and MA/Nitride fuel fabrication development for partitioning/ transmutation applications is also in progress in Europe. Nitride fuel can be recycled using the classical PUREX aqueous process. Once in solution in nitric acid the discussions of advanced aqueous recycle in L1/L2 apply and are not repeated here. In addition to the P/T work on nitride fuel recycle at JAERI, JNC has recently undertaken an intensive R&D campaign to develop pyro recycling of U/Pu/nitride fuel, with the intent to recycle TRU as a commixed product and to recover N^{15} for reuse in fuel refabrication. These pyro-based processes offer the potential for few step simple recycle and are discussed here.

Bench-scale tests have confirmed the ability to electrolytically dissolve TRU nitride fuel out of chopped segments of stainless steel cladding in a molten salt (LiCl-KCl at 773°K) electrorefiner. The ^{15}N is captured (90% recovery) for reuse in fuel fabrication. The TRU is subsequently recovered in a liquid cadmium cathode. A novel renitridation of the TRU product is being investigated wherein the ^{15}N is reintroduced directly into the liquid cadmium, to form TRU nitride.

An alternate process, LINEX, for Li_3N extraction of actinides in molten salts in also under investigation:

- Actinide nitrides are anodically dissolved in a LiCl-KCl-$CdCl_2$ salt
- The evolved nitrogen is trapped by Li, forming Li_3N
- The Li_3N is returned to the melt
- Actinides preferentially segregate as solid nitrides by reaction with Li_3N
- The alkali and alkaline earth metals and most rare earth fission products remain in the salt—facilitating separation of the refined actinide nitride from the fission products.

The Gen IV R&D plan would involve collaborative participation in these early stages of worldwide technology exploration and flow sheet development. Bench scale testing of individual process steps using non radioactive surrogates and flowsheet refinement will be done first. However because of the importance of minor actinide recycle to the achievement of Gen IV goals, Viability R&D will not be complete until tests with real minor actinides have been completed.

Beyond Viability R&D, a lengthy campaign of scaleup and prototyping (as addressed for the L1/L2 concept recycle descriptions) will be needed to carry nitride fuel recycle technology to a state of commercial readiness.

Nitrogen 15 Enrichment and Recycling R&D

The dominant isotope, ^{14}N, in natural nitrogen has a relatively large neutron capture cross-section, which results in production of the long-lived, highly mobile ^{14}C. To minimize the contribution of ^{14}C on the environmental burden from deployment of nitride-fueled reactors, isotopically-enriched (in ^{15}N) nitrogen (HE-N_2) is indispensable for nitride fuel. In order to reduce the ^{14}C production comparable with the oxide fuel, the ^{15}N enrichment (abundance) of the HE-N_2 gas is targeted as $\geq$ 99.9%.

In order to make nitride fuel economically comparable to oxide, the HE-N_2 gas cost including the initial production, recovery, and reuse in the reprocessing and fuel fabrication processes is provisionally targeted as equivalent to 10 to 20% of the fuel fabrication cost. To attain this target cost, two essential technologies have to be developed. One is an inexpensive nitrogen isotope separation technology to produce the HE-N_2 gas. The other is the minimum loss recovery technology (without the enrichment dilution) in the reprocessing and fuel fabrication based on recycle of the HE-N_2 gas.

If the cost of the HE-N_2 production is reduced enough that the once-through usage becomes less expensive than recycling, then the recovery can be eliminated. The HE-N_2 gas recovery requires additional equipment in the plant construction, and also the initial and supplements induce extra cost in the plant operation and maintenance.

At JNC, laboratory scale experiments are underway on a promising nitrogen isotope separation (the gas adsorption) technique. Also a promising nitrogen gas recovering technology, suitable for nitride fuel reprocessing and fabrication processes has been selected and its capability preliminarily evaluated. Furthermore, the nitride fuel cycle system concepts including the reprocessing and fuel fabrication process flow diagrams with the HE-N2 gas recycling have been newly designed for both aqueous and non-aqueous (pyrochemical) nitride fuel recycle plants, so that the effect of the HE-N_2 recycling on the economics of each concept can be evaluated.[i]

The Gen IV R&D program should collaboratively join in this concept evaluation. Moreover, alternative enrichment and recovery technologies should be identified and evaluated. This issue is a cost control issue which is relevant to economic but not technical viability. Even so, by the end of Viability R&D, an acceptable approach should be in evidence.

<u>Remote Refabrication R&D</u>: Aqueous

Glove box pellet fabrication technology for UN and (U, Pu) is fairly well established. However, the adaptation of this technology to highly radioactive (U, TRU) N with residual fission products, is receiving less interest than vibrocompation. And the choice of specific type vibrocompaction technology is tied closely to the choice of reprocessing technology—advanced aqueous or pyro. Since this has not be discussed in the L1/L2 refabrication writeup, it is given full coverage here.

The vibration-compaction (or vibro-packing) technique of fuel fabrication was first attempted at Oak Ridge National Laboratory (ORNL) in the U.S.A in the late 1950s. At that time, the sol-gel process was developed for Thorium Oxide preparation. At that early stage of development, the particles obtained were irregular-shaped granules (similar to shards), which induced large stresses on the cladding tubes during compaction, which would degrade the cladding. For this reason, vibration-compaction fuel pin fabrication was thought to be difficult.

Later, in the early 1960s, the gel control technique that had been developed in the catalysis industries was applied to produce well-shaped, highly dense spherical particles, and vibro-packing of spherical particles with relatively low vibration energy was attempted. In Europe, fundamental research and development of the sol-gel method was activated in 1960s. In the 1970s, the gelation process was developed, by which it became possible to produce spherical UO_2 particles with diameter larger than 600 μm, which had been difficult using the sol-gel process. Since that time, many pilot plants have been

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i. Masaki Inoue, et al., "Feasibility Study on Nitrogen-15 Enrichment and Recycling System for Innovative FR Cycle System with Nitride Fuel," Proceedings of ICONE10 10th International Conference on Nuclear Engineering, Arlington, VA, April 14-18, 2002.

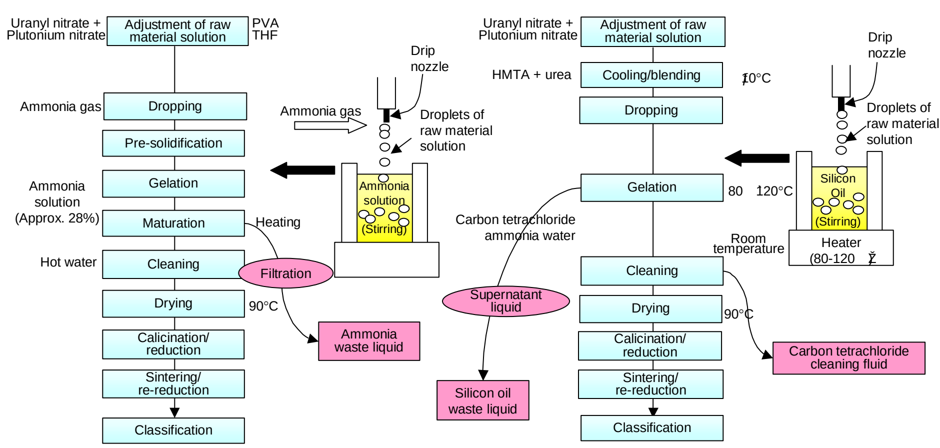
constructed and operated, and also small-scale irradiation experiments were performed in Italy, U.K., Germany and U.S.A., and elsewhere [1].

In the U.K,, irradiation experience reached about 700 pins in the Dounreay Fast Reactor (DFR) and about 3000 pins in the Prototype Fast Reactor (PFR) [2]. In the U.S.A., several pins were irradiated to burnup values as high as 12 at.% in EBR-II [3]. In Switzerland, carbide fuel was irradiated up to about 10 at.% burn-up [4]. In the late 1980s, as the development of the FBR was slowed, development activities of vibration compaction method also decreased.

In Japan, around the end of 1962, the predecessor of JNC initiated development of the sol-gel method. Development activity, however, did not reach full scale. Fundamental research was also conducted for ThO_2 fuel at the Japan Atomic Energy Research Institute (JAERI).

In Russia, development of vibration-compaction fuel fabrication combined with Oxide Electrowinning (nonaqueous recycling) started in 1970s at the Research Institute of Atomic Reactors (RIAR). In 1977, a pilot plant was constructed. Up to the present time, 426, 10 and 2 vibration-compaction MOX fuel assemblies were irradiated in BOR-60, BN-600 and BN-350, respectively, (about 18000 fuel pins in total). The maximum burnup attained in the BOR-60 with test pins of this type of fuel was about 32 at.% [5]. RIAR also has experience with production of granules using vapor oxidation of fluorides [6].

Gelation processes for aqueous recycling is categorized as either external gelation or internal gelation. In both cases, the process sequence is similar. Namely, a specific organic agent is added first to an aqueous solution containing heavy metals. Then, the solution is dropped as a droplet into another solution or bath. In the course of this process, the droplets become a spherical gel. In the Russian process applied to oxide-electrowinning, U and Pu dissolved in chloride salt solution are precipitated to an electrode as oxides, and such oxide granules are subsequently crushed and compacted into a fuel rod. Figure 5 provides a schematic of the two-gelation processes.



Granule production flow by external gelation

Granule production flow by internal gelation

Particle Fabrication that Matches with Aqueous Reprocessing

Figure 5. Schematic illustration of external and internal gelation processes.

The vibration technologies are categorized into following as either electrical vibration, pneumatic vibration. or electromagnetic vibration. The electromagnetic vibration method is applied to internal gelation at the Paul Scherer Institute (PSI) in Switzerland and to production of vibro-pack fuel in Russia using oxide electrowinning product. Figure 6 is a schematic illustration of the incorporation of vibration-compaction fuel fabrication into each of the various recycling schemes.

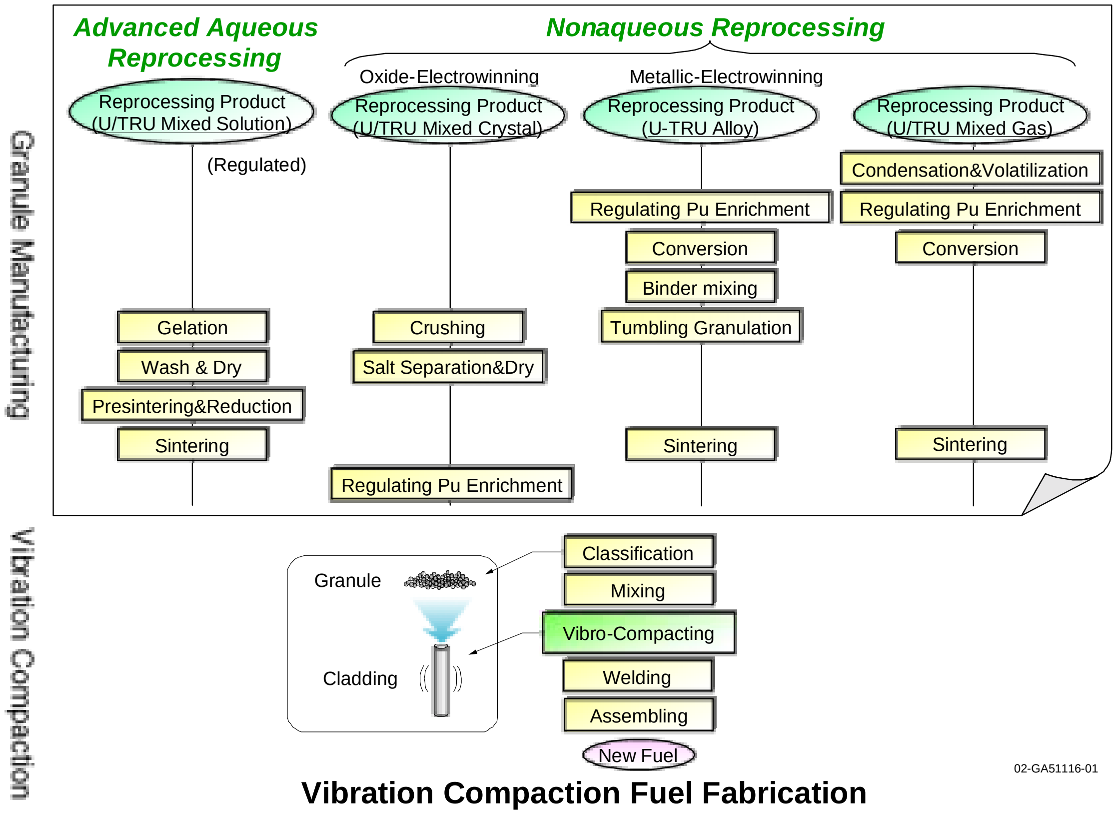


Figure 6. Incorporation of vibration-compaction fuel fabrication into various recycling methods.

For the case of spherical, compacted fuel, the packing density is determined geometrically by the diametrical ratio of spherical particles and by the numbers of diameter sizes of particles. On the other hand, for the case of irregular-shaped granules, the edges of granules are ground by the vibration energy, which makes packing density higher.

The main development need for vibro-compacted fuel with spherical particles is the improvement of the smear density. For vibro-compacted fuel with irregular-shaped particles, it is important to find the vibration condition that minimizes the scars on the inner surface of cladding.

It is said that remote operation would be easily attained for sphere compacted fuel fabrication because the process does not include powder-handling steps, and as a result there will be no dispersion of fine powders.

Because the solution material of external gelation is thermally stable, the process of external gelation is considered to be simple and efficient. On the other hand, the amount of waste solution is produced largely and the sphere is likely to form a shell structure. Granules produced by the internal gelation method have a high degree of sphere-like shape and homogeneity; however, the solution material is thermally unstable such that appropriate cooling equipment is required. Further, the processing of used silicon-oil bath will be needed.

Vibration-compaction fuel fabrication by pyroelectrochemical processing is characterized by a small number of steps due to its combination with recycling, such that it would be suitable to a scale-up of the facility. On the other hand, there is a possibility that powder may be dispersed in the fabrication hot cell when granules are crushed, increasing contamination. The use of chlorine for the dissolution of materials is taken as one of the inferior features of this method. In this process, oxygen getter made of metallic Uranium powders is used in order to control the oxygen potential in fuel rods.

Currently, technological development of vibration-compaction is underway in Switzerland, Russia and Japan etc. at the moment.

At PSI in Switzerland, research on vibration-compaction with spherical particles derived using the internal gelation method (termed “sphere-pack” fuel fabrication) is begin performed. In Japan, the JNC started research on gelation in 1990, and is presently proceeding to the collaborative study on fuel fabrication with PSI. At RIAR in Russia, irregular-shaped vipac fuel fabrication by pyroelectrochemical method and its irradiation in BOR-60 is being continued.

With regard to the gelation method, the establishment of the optimum condition of gelation and treatment of waste solution remains to be solved. The means of solving such items and more efficient granulation method are to be developed. In the case of low-decontaminated and TRU fuel, it will be necessary to show the applicability of gelation to the multicomponent systems.

Regarding the Oxide Electrowinning method, the Pu enrichment distribution throughout the fabrication process should be monitored with quality control.

Development tasks that are common to both aqueous and non-aqueous recycling are optimization of vibration conditions in order to attain high-density fuel, a non-destructive inspection method for low decontaminated fuel, and irradiation experiments to confirm good irradiation performance.

Remote Refabrication R&D: Pyro

TRU-Nitride feedstock particles for vibrocompaction will be made either directly by crushing the shards generated in pyro recycle such as the electrorefining or LINEX processes described in the former section; or by carbothermic reduction of actinide oxide particles already made by gelatin techniques following aqueous recycle.

A program of work on fabrication and subsequent property testing of TRU and of specific minor actinide nitrides is underway at JAERI. The Gen IV Viability R&D program would join collaboratively in this work to achieve a level of confidence in ability to produce TRU nitride pins of consistent morphology by remote vibrocompaction refabrication techniques.

Especially important to viability is verification that losses to waste can be avoided without reliance on costly secondary waste cleanup processes.

Property Measurements R&D

Although considerable work has been performed in Japan in recent years the phase equilibria and thermophysical properties of (U,TRU)N compounds are not yet fully known. Further work is needed on:

- Assessment of phase equilibria of TRU nitrides at temperatures of interest, with particular attention to the affect of minor actinide additions on dissociation temperatures
- Thermophysical property measurement
- Confirmation of fuel compatibility and coolant with minor actinide additions.

Irradiation Performance R&D

The irradiation performance database for (U,Pu)N is small, but sufficient to indicate that utilization of (U,Pu)N to moderate burnups (about 10 to 12 at.% burnup) may be possible. However, there is little or no experience with (U,TRU)N compositions and no experience that would allow assessment of limiting conditions of operation. An irradiation testing program, similar to that described for Viability R&D with recycled U-Pu-Zr is envisioned.

- Assessment of (U,Pu)N behavior and lifetime potential over the full range of nominal and 2-sigma operating conditions.
- Assessment of transient overpower and loss-of-flow behavior of (U,Pu)N at various stages of burnup, with particular attention to behavior of fuel and retained fission gas and of dissociation effects (if any) during transient overpower events.
- Assessment of the impact of minor actinide contents to the steady-state and transient behavior of (U,Pu)N.
- Development of cladding alloys (or identification and evaluation of existing alloys) with improved high-temperature strength over HT9, but with similar resistance to swelling.

Modeling and Code Development R&D

No validated code is available for predicting performance of nitride fuel. However, life-limiting phenomena for such fuel at higher burnup values (15 to 20 at.%) have not been identified. Work is underway under NERI funding on an L6 concept to develop a nitride fuel pin modeling code. The achievement of a fully validated fuel performance code must be deferred to a post-Viability R&D phase when a robust irradiation database has become available. The issues related to viability are limited to the development of models to allow an understanding of any life-limiting phenomena.

- Development of mechanistic models for life-limiting phenomena and other performance characteristics as they are identified, or adaptation of those developed for $(U,Pu)O_2$, if appropriate.
- Understand dissociation phenomenology of nitride at high temperature

Thorium-Based Alloy Fuel R&D Plan

Concept set L4 proposes thorium-based fuel for a waste management mission. The thorium is substituted for U-238 to avoid in-situ generation of new transuranics, for a conceivable future in which TRU or at least Minor Actinides are viewed as a waste. The thorium is introduced for neutronics and safety reasons, and although a specific fuel type was proposed (Th/U/TRU/Zr alloy), the focus was on neutronics, and not on fuel/clad/coolant performance nor recycle performance.

For that reason, the initial step in the thorium based fuel R&D plan will be to survey the compositions and fuel forms (metal, alloy, met-met dispersion, oxide, and nitride) regarding both irradiation performance and recycle methods, before launching a directed R&D program.

Survey of Thorium-Based Fuel Experience Base

Thorium-based fuel received significant research in the early decades of the nuclear era including an extensive program for the Shippingport Light Water Breeder.

Experience was accumulated on thorium oxide fuel in pin geometry and thorium-based particulate fuel for gas cooled thermal reactors. Thorium fuel cycles for molten salt liquid fueled reactors was also studied extensively in the 50s and 60s. Argonne performed Th alloy based fuel testing in the early decades. Recently work on thorium based fuels has been for LWR service as a potential way to extend fuel burnup.

This history has been surveyed from time to time, most recently by the Gen IV Fuel Cycle Crosscut Group.

Recycling technology (THOREX) was to the point that a pilot plant was built in Italy. Molten Salt Reactor recycle was integrated in the program at Oak Ridge. Thorium fuel processing was performed in the Light Water Breeder Program to assess breeding.

For the L4 actinide management mission development, the first stage of the Th-based fuel R&D program will be to go over the accumulated experience base on different fuel forms and compositions to assess: (1) fabricability, (2) irradiation performance, and (3) recycleability to confirm the tentative selection of an alloy fuel form and pyro-based recycle/injection casting based refabrication.

Moreover options for the beneficial use of the U233 generated in the L4 actinide management mission will be evaluated and refabrication and recycle or waste management technology options will be screened and downselected. (It is anticipated that the downselected options will involve U233 recycle back to thermal reactors). On that basis the recycle/refab of U233 based fuels will not be discussed here—assuming that they will be covered in the appropriate Technical Working Group (TWG-1, 2, or 4).

Th/U/MA/Zr Recycle

If any alloy fuel is selected as a result of the screening, it is anticipated that this fuel would be recycled using pyro technology—which is already partially developed, within the AAA program, for U/Pu/MA/Zr dispersion fuel. For that fuel, flowsheets have been developed which first "digest" the Zr matrix from the dispersion fuel, and then subject the U/Pu/MA material to the pyrometallurgical recycle technology already substantially developed for the IFR.

The introduction of the Thorium can be anticipated to give rise to new issues and challenges – which the survey will uncover.

Th/U/Pu/MA/Zr Fuel Refabrication R&D

If an alloy fuel is selected as a result of screening, it is anticipated that this fuel would be fabricated along the same lines as the U/Pu/MA/Ar dispersion fuel under development in the AAA program.

Powder metallurgy sintering technology might be used to form a dispersion fuel form. This strategy would be considered to avoid temperatures which would be so high as to volatilize the Americium in the mix of minor actinides during fabrication.

Materials Properties R&D

The physical/chemical/thermal properties of the as fabricated fuel form will be determined over the full temperature span of operational conditions. This will be done in an iterative process as the fabrication techniques are developed and refined. Testing will start with unirradiated materials and surrogate recycle materials (e.g., U238 may be used instead of U233 to avoid its U232 radioactive contaminant). Properties determination of irradiated material will be part of the post irradiation test program.

Irradiation Performance R&D

The sequencing of the irradiation testing program and its integration with the recycle/refabrication development program will progress as discussed above for the nitride fuels program. The two programs would if necessary benefit by sharing the same facilities and in particular the same test reactor and associated post-test examination hot cells and diagnostic equipment.

Safety Testing R&D

The safety testing R&D program will follow a similar sequence to that already discussed for the nitride fuel development. Again the same facilities could be used for both fuels.

4.4.10 Energy Conversion R&D

Technology Gaps

As indicated in Table 2, the L6/L4 concept sets include proposals to move away from traditional superheated Rankine cycles—motivated by two goals

- To improve cost competitiveness of electricity generation
- To expand the menu of energy services that nuclear supplied heat can provide.

In the first area, both supercritical steam Rankine cycles (27 Mpa) and supercritical CO_2 Brayton cycles (20 Mpa) are proposed—each producing conversion efficiencies near 45% with a dramatically reduced BOP footprint and scale of equipment size and/or complexity. Both can use invessel IHX's because of the chemical inertness of Pb with H_2O or CO_2. The supercritical CO_2 Brayton cycle is immensely smaller and less complex than a Rankine cycle and would appear to hold exceptional potential for cost reduction—both capital and O&M. Supercritical steam allows reduction of turbine complexity by reducing the fraction of stages that handle wet steam. The R&D needed is modest compared to the potential payoff; in the case of supercritical steam cycles, it is synergistic with the TWG-1 concepts.

Desalinization bottoming cycles take little away from electricity production (accepting heat at no higher than 125ºC), and small plants have been coupled to numerous LWR's in Japan already (Diablo Canyon in the US). The market for potable water as a revenue source is projected to grow during the time period of Gen IV commercialization. The technology is off the shelf, but various optimization strategies for combining Reverse Osmosis with thermal methods are still being refined.

Process heat applications of nuclear heat exist at every temperature range: 325°C to > 1000°C. Recycle paper mills are a very large user of low heat steam—and as gas prices increase could provide a potential market for LWR plants. At the 550°C of sodium-cooled plants (L1/L2), technologies for sorption-enhanced or membrane-based nuclear assisted steam—methane reforming production of hydrogen open up. Finally, certain of the L6 concepts propose to exploit the high boiling temperature of Pb coolant so as to reach the 750 to 800°C range needed for thermochemical water cracking manufacture of hydrogen. (Much of the high temperature materials R&D enumerated in previous sections derives from the motivation to achieve 800°C core outlet temperature for the hydrogen manufacture mission.)

R&D is needed for the thermochemical process itself and for the high temperature materials to contain the water cracking reagents.

Coupling a nuclear heat source to process heat applications implies siting near the population centers supplying the process heat workforce. This gives rise to R&D directed to achieving unprecedented levels of safety.

Payoffs

The area of energy conversion innovation captured especially in the L6/L4 concepts of Gen IV are areas of especially high potential payoff. The Rankine cycle BOP equipment contribute as much to capital cost as does the NSSS itself. Rankine steam cycle development reached its peak in the 1950s. Subsequently the Brayton cycle has achieved dominance in new plant construction—benefiting from 50 years of military aerospace spending on gas turbines. Moving from a Rankine to a much simpler and smaller Brayton cycle BOP holds a potential for major cost savings in liquid metal cooled systems.

Broadening the energy product mix available from nuclear also holds major payoff potential. Electricity comprises only 1/3 of energy use of society; the rest is delivered as heat supplied by fossil fuels. By manufacturing hydrogen by methane splitting—and ultimately by water splitting, nuclear could enter the additional 2/3 of the market where it is currently excluded. Again the potential payoff is very large.

Process heat applications may become important as fossil reserves drive their prices up or as pressure builds to reduce CO_2 emissions.

Finally, water production is a crucial element of sustainability in all parts of the world—developed and developing, and the potential payoff of desalination is very large.

Potential Showstoppers

The supercritical CO_2 cycle can be deployed at traditional fast reactor temperatures of 550°C core outlet temperature; there appear to be no showstoppers. Thermochemcial water cracking will likely require 800°C core outlet temperature; here a potential showstopper is the need for development of suitable structural and cladding materials which can endure the service conditions not only in the reactor but also in the chemical plant—and to be fabricable at acceptable cost. The thermochemical water cracking R&D will also depend on the selection of suitable materials of construction for the high temperature service conditions in corrosive fluids.

R&D Plan

Supercritical CO_2 Brayton Cycle R&D

The thermophysical properties of supercritical CO_2 (large Cp at high pressure but small Cp at lower pressure) complicate the thermodynamic cycle optimization and the recuperator design. Cycle optimization is where the work should start. Material selection for components comes next, and finally small scale testing of turbine stages and recuperator segments is needed to establish viability.

Supercritical Steam Rankine Cycle R&D

This work is a focus area for TWG-1. Here it would be required only to monitor their work on BOP components; the relevant structural and safety issues for L6 concepts was discussed previously.

Ca-Br Thermochemical Water Cracking R&D

The Ca-Br cycle operates at ~750°C and sets the target 800°C outlet temperature goal for Pb cooled reactors in concept set L6. The cycle itself requires extensive R&D (as do all others). They include selection of materials for heat exchangers and for chemical process vessels and piping; development of a Ca support substrate for the reactant; properties measurements of all reactants and reaction products; and rate constant measurements. Then a flow sheet can be optimized iteratively with bench scale testing. Finally a prototype plant should be designed built and tested.

Desalinization R&D

The basic technologies are already in existence. However for the viability phase of Gen IV development, the work is primarily optimization and economics tradeoff studies to select optimal nuclear heat allocations for a combined cycle plant producing both electricity and potable water and maximizing profitability.

5. REFERENCES

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[5] V. B. Ivanov et al., Proc. Int. Conf. Future Nuclear Systems – Challenge Towards Second Nuclear Era with Advanced Fuel Cycles, Oct. 5–10, 1997, Yokohama, Japan (1997) vol. II, p.1093 (Global '97).

[6] V. B. Ivanov et al., ibid, p.1099.

Appendix A

R&D Summary

TWG 3 (L1/L2)
R&D Scope for Section 3.1.1 (Advanced aqueous process and remote fabrication)
Version 073102

	Technical gap/issue				R&D Items				
Sub-System	Gap Label	Brief Description of Gap/Issue	Signific. of Gap *(a)*	Current TRL *(b)*	Activity Label	Brief Description of R&D Activity	Priority *(c)*	Time *(d)*	Estimated Cost Range (Million USD) *(e)*
Fuel Cycle	FCA1	Development of integrated disassembling/shearing system	O	4	FCA1a	Improve laser performance and endurance of optical fiber	3	M	5-10
					FCA1b	Develop a compact disassembly/shearing machine with laser	3	M	
	FCA2	Development of decladding system	P	3	FCA2a	Demonstrate a shredding equipment by an engineering-scale cold (uranium) test	3	M	
	FCA3	(Crystalization process) Development of uranyl nitrate hexahydrate (UNH) crystallization technology	P	4	FCA3a	Develop UNH crystallization technology with enough DF for TRU and FPs by small-scale hot tests	2	S	5-10
	FCA4	(U/Pu/Np co-recovery) Development of advanced solvent extraction process	O	5	FCA4a	Optimize extraction condition by small-scale hot tests	3	S	
	FCA5	(MA recovery) Development of advanced MA recover technology to minimize HLW	P	3	FCA5a	Develop a salt-free MA recovery technology with high separation capability from lanthanides by small-scale hot tests	2	S	
	FCA6	Demonstrate main equipments by engineering-scale cold (uranium) tests	P	4	FCA6a	Dissolver	3	M	5-10
					FCA6b	Crystallization equipment	3	M	
					FCA6c	centrifugal contactor system	3	M	
	FCA7	Minimal experience for low decontamination MA-bearing pelletizing technology	V	3	FCA7a	Develop a denitration/conversion equipment adequate for simplified system by small-scale hot tests	2	M	10-20
					FCA7b	Verify simplified pelletizing process for low decontamination MA-bearing fuel by small-scale hot tests	1	S	
					FCA7c	Develop ODS cladding welding/inspection equipment	2	M	
					FCA7d	Assess and demonstrate fuel fabrication equipment (quality / process control equipment) by an engineering-scale cold (uranium) test	2	M	
	FCA8	Development of remote maintainable equipment	P	3	FCA6a	Develop remote maintenance equipment	2	M	10-20
					FCA6b	Assess and demonstrate remote maintenance technology by an engineering-scale cold (uranium) test	3	M	
	FCA9	Minimal experience for vibropacked fuel fabrication technology option	V	2	FCA7a	Develop gelation process equipment by small-scale hot tests	1	S	5-10
					FCA7b	Demonstrate gelation system by an engineering-scale cold (uranium) test	2	M	
	FCA10	(High Level Waste) Increase FP content to about 30% by removing Mo, Sr and Cs	P	3	FCA8a	Assess and demonstrate high-loading vitrification process	3	M	

Sub-System	Technical gap/issue				R&D Items				
	Gap Label	Brief Description of Gap/Issue	Signific. of Gap *(a)*	Current TRL *(b)*	Activity Label	Brief Description of R&D Activity	Priority *(c)*	Time *(d)*	Estimated Cost Range (Million USD) *(e)*
	FCA11	(TRU waste) Reduce TRU waste	O	4	FCA9a	Assess and demonstrate decontamination technology	3	M	2-5

a *Indicate relevance of technology gap: V= concept viability, P = performance, O = design optimization*

Sum: $42-85 M

b *Indicate technical readiness level (1, 2, 3, 4, or 5); see EMG Final Screening Document*

c *Indicate priority of R&D activity:*

1 = critical (needed to resolve a key feasibility or viability issue)

2 = essential (needed to reach a minimum targeted level of performance, or to resolve key technology or performance uncertainties)

3 = important (needed to enhance performance or resolve the choice between viable technical options)

d *Indicate time required to perform R&D: S = short (<2y), M = medium (2-5y), L = long (5-10y), VL = very long (>10y)*

e *Indicate costs for test equipment and materials*

The personnel expenses, facility running costs (utilities and maintenance costs) and test building construction costs are excluded in the estimated cost range

f *The R&D items in engineering-scale hot tests are excluded in this table*

TWG 3 (L1/L2)
R&D Scope for Section 3.1.2 (Pyroprocess)
Version 073102

	Technical gap/issue				R&D Items				
Sub-System	Gap Label	Brief Description of Gap Issue	Signific of Gap *(a)*	Current TRL *(b)*	Activity Label	Brief Description of R&D Activity	Priority *(c)*	Time *(d)*	Estimated Cost Range (Million USD) *(g)*
Fuel Cycle	FC1	Parameters to scale from laboratory tests to the engineering scale are needed	V	2	FC1a	Complete laboratory-scale testing with surrogates	1	S	15-20
					FC1b	Perform scale-up testing with surrogates	1	S	
					FC1c	Perform laboratory-scale tests with irradiated materials	1	S	
					FC1d	Perform prototype demonstration with irradiated materials	2	M	
					FC1e	Perform integrated demonstration with irradiated materials	2	M	
	FC2	Size reduction (shredding) may be required for oxide fuels	P	3	FC2a	Test improved shearing systems for oxide fuels	2	M	2-4
					FC2b	Perform demonstration of oxide shearing system with irradiated fuel	2	M	
	FC3	Chopping experience for metal fuels limited to single pins	O	5	FC3a	Assess fuel shearing requirements for throughput	3	S	1
					FC3b	Test improved shearing systems for metal fuels	3	M	
	FC4	Minimal experience exists in electrorefining reduced oxide	P	3	FC4a	Test electrorefining system with reduced oxides	2	M	3-5
						Electrorefine reduced oxides in integrated demonstration			
	FC5	Increased electrorefing in throughputs may be required	P	3	FC5a	Test high throughput systems with surrogates	2	S	8-10
					FC5b	Test high throughput systems with irradiated materials	2	M	
					FC5c	Perform integrated demonstration with irradiated materials	2	M	
	FC6	Recovery processes for transuranics, including plutonium, need to be demonstrated	V	3	FC6a	Test laboratory-scale TRU recovery	1	S	8-10
					FC6b	Perform prototype demonstration of TRU recovery	1	S	
					FC6c	Perform integrated demonstration with irradiated materials	2	M	
	FC7	Minimal experience with drawdown equipment for actinide removal from salts for waste processing	V	2	FC7a	Test drawdown systems with surrogates	2	S	4-6
					FC7b	Test drawdown systems with irradiated materials	2	M	
	FC8	Improvements may be needed in salt separation systems	P	4	FC8a	Perform optimization tests on salt separation system	2	S	8-10
					FC8b	Test improved salt separation systems with irradiated materials	3	M	
	FC9	Improvements may be needed for metal fuel casting	P	4	FC9a	Perform optimization tests on casting technologies	3	S	3-5
					FC9b	Test improved pin mold technologies	2	M	
					FC9c	Demonstrate improved pin casting system	3	M	
	FC10	New materials to minimize losses are needed for high temperature casting operations	P	3	FC10a	Test improved materials for fuel casting	3	M	6-10
					FC10b	Test improved materials for salt separation	3	M	

	Technical gap/issue				R&D Items				
Sub-System	Gap Label	Brief Description of Gap Issue	Signific of Gap *(a)*	Current TRL *(b)*	Activity Label	Brief Description of R&D Activity	Priority *(c)*	Time *(d)*	Estimated Cost Range (Million USD) *(g)*
	FC11	Improvements may be needed in remote fabrication technologies	O	5	FC11a	Perform review of remote fabrication experience	3	S	3-5
					FC11b	Test remote fabrication systems	3	M	
	FC12	Qualification of metal HLW needs to be completed	P	4	FC12a	Assess metal waste options for oxide fuels	2	S	4-6
					FC12b	Complete metal waste qualification testing	2	M	
					FC12c	Prepare waste qualification documentation and QA system	2	M	
					FC12d	Complete process qualification of metal waste production system	2	M	
	FC13	Qualification of ceramic HLW needs to be completed	P	4	FC13a	Complete ceramic waste qualification testing	2	M	4-6
					FC13b	Complete process qualification of ceramic waste production system	2	M	
	FC14	Minimal experience exists with ion exchange systems for reducing ceramic waste volume	V	2	FC14a	Assess impacts of ion exchange on waste volumes	2	S	8-10
					FC14b	Perform laboratory-scale test of ion exchange systems	2	S	
					FC14c	Perform prototype testing of ion exchange systems with surrogates	2	M	
					FC14d	Perform prototype testing of ion exchange systems with irradiated	2	M	
	FC15	Final process models need to be developed	O	5	FC15a	Complete development of process models for unit operations	3	S	1
					FC15b	Validate process models	3	S	
	FC16	Improvements may be needed in the accountability system	P	4	FC16a	Assess present status of accountability system	2	S	1
					FC16b	Develop improvements to accountability system	3	S	
	FC17	An assessment of the transparency of the pyro fuel cycle may be needed	P	4	FC17a	Assess transparency of the pyro fuel cycle	2	S	1
					FC17b	Demonstration transparency of the pyro fuel cycle	3	M	
	FC18	NDA techniques may be needed to assay process streams	P	4	FC18a	Develop NDA techniques for process streams including fuel	2	M	3-5
					FC18b	Demonstrate NDA techniques with process materials	2	M	

See the first table in this Appendix for explanation of the footnote legend

Footnote (e) does ***not*** *apply*

g Costs for hot integrated demonstration at engineering scale are excluded.

TWG 3 (L1/L2)
R&D Scope for Section 3.1.3 (Other dry processes and Vibropac fuel fabrication)
Version 073102

Sub-System	Technical gap/issue				R&D Items				
	Gap Label	Brief Description of Gap/Issue	Signific. of Gap *(a)*	Current TRL *(b)*	Activity Label	Brief Description of R&D Activity	Priority *(c)*	Time *(d)*	Estimated Cost Range (Million USD) *(e)*
Fuel Cycle	FC01	Development of integrated disassembling/shearing system	O	4	FCO1a	Improve laser performance and endurance of optical fiber	3	S	5-10
					FCO1b	Develop a compact laser disassembly/shearing machine	3	M	
	FCO2	Verification of decladding and fuel powdering technology	V	2	FCO2a	Verify powdering performance of spent fuel, separation efficiency of powdered claddings from spent fuel, and high fuel recovery	1	M	
					FCO2b	Demonstrate shredding equipment by an engineering-scale cold (uranium) test	2	M	
	FCO3	Development of anode dissolution and chlorinating dissolution processes	P	3	FCO3a	Test fuel dissolution properties of U and Pu by small-scale hot tests	2	S	2-5
					FCO3b	Demonstrate fuel dissolution process	2	M	
	FCO4	Development of noble metals removal process	P	3	FCO4a	Test noble metals separation behavior by small-scale hot tests	2	S	
					FCO4b	Demonstrate noble metals removal process	2	M	
	FCO5	Verification of MOX co-deposition process	V	2	FCO5a	Verify MOX co-deposition properties by small-scale hot tests	1	S	10-20
					FCO5b	Demonstrate MOX co-deposition process	1	M	
	FCO6	Verification of MA recovery process	V	2	FCO6a	Verify MA recovery technology by small-scale hot tests	1	S	
					FCO6b	Demonstrate MA recovery process	1	M	
	FCO7	Development of electrowinning equipment	P	3	FCO7a	Test salt heating/cooling performance and volatile salt behavior	2	M	5-10
					FCO7b	Evaluate handling technology under high temperature and corrosion resistance of materials	2	M	
					FCO7c	Demonstrate electrowinning equipments with high corrosion resistance	3	M	
	FCO8	Development of chlorine recycle system	P	3	FCO8a	Improve chlorine recovery performance	2	M	2-5
					FCO8b	Demonstrate chlorine recycle system	3	M	
	FCO9	Development of cathode treatment system	P	3	FCO9a	Improve recovery performance	2	M	
					FCO9b	Demonstrate cathode treatment system	3	M	
	FCO10	Development of salt separation system	P	3	FCO10a	Evaluate vacuum distillation performance	2	S	
					FCO10b	Demonstrate distillation components	3	M	
	FCO11	Verification of fuel meat preparation route	V	2	FCO11a	Classify suitable distribution of particle diameters for the vibropac process	1	M	

Sub-System	Technical gap/issue				R&D Items				
	Gap Label	Brief Description of Gap/Issue	Signific. of Gap *(a)*	Current TRL *(b)*	Activity Label	Brief Description of R&D Activity	Priority *(c)*	Time *(d)*	Estimated Cost Range (Million USD) *(e)*
	FCO12	Development of fuel fabrication technology	p	3	FCO12a	Assess operation condition range to satisfy the fuel specification	2	S	10-20
					FCO12b	Assess and demonstrate fuel fabrication equipment (quality/process control equipments) by engineering-scale cold (uranium) tests	2	M	
	FCO13	Development of remotely maintainable equipment	P	3	FCO13a	Develop remote maintenance equipment	2	M	10-20
					FCO13b	Assess remote maintenance technology by an engineering-scale cold (uranium) test	3	M	
	FCO14	(High-level waste) Development of vitrification process for the phosphate precipitations including fission products and the waste salts	P	3	FCO14a	Test decontamination efficiency in the phosphate precipitation process by cold precipitation tests	2	M	2-5
					FCO14b	Improve oxidation efficiency of waste salts	2	M	
					FCO14c	Demonstrate vitrification process	2	M	
	FCO15	(TRU waste) Reduce TRU waste	O	4	FCO15a	Assess and demonstrate decontamination technology	3	M	
	FCO16	Development of process control and accountability system	P	3	FCO16a	Develop and demonstrate analysis methods and analysis system	2	M	2-5
					FCO16b	Construct safeguards concept	2	M	
					FCO16c	Develop and demonstrate in-situ analysis method including sampling	3	M	
	FCO17	Development of nuclear criticality safety management system	P	3	FCO17a	Develop and demonstrate nuclear criticality safety management system	3	M	1-2

Sum: $42-102 M
See the first table in this Appendix for explanation of the footnote legend
Footnotes (e) and (f) **do** apply

TWG 3 (L1/L2)
R&D Scope for Section 3.2 (Safety)
Version 073102

	Technical gap/issue				R&D Items				
Sub-System	Gap Label	Brief Description of Gap/Issue	Signific. of Gap *(a)*	Current TRL *(b)*	Activity Label	Brief Description of R&D Activity	Priority *(c)*	Time *(d)*	Estimated Cost Range (Million USD) *(e)*
Safety	S1.1	Establish pre-failure fuel behavior in overpower accidents to verify passive safety phenomena	P	3	S1.1a	Design and perform new TREAT experiments to fill data gaps	2	M-L	20-50
					S1.1b	Utilize data to validate code predictions for DBAs and ATWS	2	M-L	1-2
	S1.2	Provide a data base to evaluate fuel failure and fission product release from metal fuels	P	2	S1.2a	Design and perform new TREAT experiments to fill data gaps (experiments may be extensions of S1.1a)	2	M-L	incl. in S1.1a
					S1.2b	Utilize data to validate code predictions and to evaluate radiological release issues	2	M-L	incl. in S1.1b
	S1.3	Provide a data base to evaluate post-accident coolability of damaged fuel	P	2	S1.3a	Examine experiments of S1.2a for applicable data	2	M-L	0.5-1
					S1.3b	Design and perform new CAMEL-type experiments to verify coolability of damaged fuel.	2	M	1-2
	S2.1	Verification of the predictability and effectiveness of radial core expansion, subassembly bowing and control rod driveline expansion as reactivity feedback mechanisms in undercooling accidents	P	4	S2.1a	Evaluate need for and feasibility of experiments to validate predictions of radial core expansion, subassembly bowing, and control rod driveline expansion	2	S	1-2
					S2.1b	Design and perform experiments judged needed and feasible in S2.1a	2	M-L	TBD
	S2.2	Establish pre-failure behaviour, fuel failure and fission product release for metal fuel in undercooling accidents	P	2	S2.2a	Design and perform TREAT experiments to investigate pre-failure fuel behavior, fuel failure, and fission product release in undercooling accidents	2	M-L	20-50
					S2.2.b	Utilize data to validate code predictions and to evaluate radiological release issues	2	M-L	1-2
	S2.3	Establish coolability of metal fuel after failures in an undercooling accident	P	2	S2.3a	Evaluate data from S2.2a experiments to assess coolability of damaged fuel	2	M-L	incl. in S2.2b
					S2.3b	Design and conduct out-of-pile CAMEL-type experiments to evaluate coolability of damaged fuel	2	M	1-2
	S3.1	Establish long-term coolability of oxide and metal fuel debris after a bounding case accident	V	2	S3.1a	Design and conduct out-of-pile experiments to characterize oxide and metal fuel debris beds and evaluate their coolability	1	M	5-10
					S3.1b	Evaluate data from S1.2b and S2.2a experiments to characterize fuel relocation and debris properties	2	M	1-2

See the first table in this Appendix for explanation of the footnote legend

Footnotes (e) and (f) do <u>not</u> apply

TWG 3 (L1/L2)

R&D Scope for Section 3.2 (Safety)

Version 073102

	Technical gap/issue				R&D Items				
Sub-System	Gap Label	Brief Description of Gap/Issue	Signific. of Gap *(a)*	Current TRL *(b)*	Activity Label	Brief Description of R&D Activity	Priority *(c)*	Time *(d)*	Estimated Cost Range (Million USD) *(e)*
Safety Facilities	S4.1	Demonstration of the effectiveness of the SASS system	P	3	S4.1a	Testing of candidate alloys for thermal aging, thermal fatigue, creep, thermal transients, and radiation effects	2	M	5-10
					S4.1b	Testing of a full scale SASS system in a suitable sodium loop with transient capability	2	M-L	incl. in S4.1a
					S4.1c	In-pile test to demonstrate the integrity of SASS under irradiation conditions	2	M-L	2-5
	S6.1	Establish in-vessel debris retention for metal fuel	V	2	S6.1a	Review relevant experiments and evaluate data from activities S3.1a and other data as available	1	M	1-2
					S6.1b	Design and perform additional out-of-pile experiments as required	1	M-L	TBD
	S6.2	Provide experimental evidence that molten metal fuel will drain from the core region to preclude recriticality	V	2	S6.2a	Perform analysis, modeling, and code development to evaluate fuel removal by draining	1	M	1-2
					S6.2b	Design and perform suitable out-of-pile experiments using molten metal and inlet assembly mockups for code validation	1	M	1-2
					S6.2c	Review data on melt spreading and debris bed formation to evaluate the potential for recriticality	1	M	1-2
					S6.2d	Design and conduct additional out-of-pile experiments on melt spreading and debris bed formation as required.	1	M	TBD
	S7.1	Provide advanced thermal-hydraulic-neutronic-structural code capability to assess passive response to DBAs and ATWS events	P	4	S7.1a	Code development activities based on the SAS code	2	M	2-5
					S7.1b	Code validation activities based on data from S1.1, S1.2, S2.1, S2.2	2	M-L	2-5
	S7.2	Provide extended modeling to assess fuel failure and fuel relocation in bounding events	P	3	S7.2a	Extend SAS metal fuel models to include fuel failure and fuel relocation	2	M	incl. in S7.1a
					S7.2b	Code validation activities	2	M-L	incl. in S7.1b
	S8.1	Development of "recriticality free" design features for oxide-fueled systems	V	3	S8.1a	Demonstration of the molten fuel discharge capability of the FAIDUS concept	1	M	5-10
					S8.1b	Demonstration of the molten fuel discharge capability of the ABLE concept	3	L	incl. in S8.1a
	SF1.1	Availability of required in-pile experimental facilities	P	N/A	SF1.1a	Restart of TREAT and preparation for HFEF test support	2	S-M	10-20
	SF1.2	Availability of required out-of-pile experimental facilities	V	N/A	SF1.2a	Provide out-of-pile facilities for experiments including fuel, sodium	1	S-M	5-10

See the first table in this Appendix for explanation of the footnote legend

Footnotes (e) and (f) do **<u>not</u>** apply

TWG 3 (L1/L2)
R&D Scope for Section 3.3 (Fuels)
Version 073102

	Technical gap/issue				R&D Items				
Sub-System	Gap Label	Brief Description of Gap/Issue	Signific. of Gap *(a)*	Current TRL *(b)*	Activity Label	Brief Description of R&D Activity	Priority *(c)*	Time *(d)*	Estimated Cost Range (Million USD) *(e)*
Fuel	F.M.1	Property Determination							
(Metal Alloy)	F.M.1.1	Confirmation of key thermophysical properties for minor actinide-bearing fuel	V	2	F.M.1.1	Measurements of thermal conductivity, heat capacity, and thermal expansion of recycled U-Pu-Zr (with minor actnide additions) at key conditions to compare to those of U-Pu-Zr	2	S	2
	F.M.1.2	Thorough evaluation of temperature-dependent thermopphysical properties	P	2		Measurement of temperature-dependent thermophysical properties of U-Pu-Zr (with minor actinide additions)	2	M	4
	F.M.1.3	Confirmation of fuel/cladding constituent interdiffusion behavior for minor actinide-bearing fuel	V	2		Diffusion couple tests to compare fuel/cladding interdiffusion behavior (with minor actinide additions) to that of U-Pu-Zr	2	S	2
	F.M.1.4	Thorough assessment and modeling of fuel/cladding interdiffusion behavior	P	2		Assessment of fuel/cladding interdiffusion behavior and development of a predictive model	2	M	4
	F.M.1.5	Thorough assessment of phase equilbria in the U-Pu-Zr system with minor actinide additions	P	2		Phase identification and stability determination for important temperature and composition ranges	3	M	4
	F.M.2	Irradiation Performance		3					
	F.M.2.1	Irradiation performance of fuel with minor actinide additions	P	3	F.M.2.1a	Steady state irradiation testing of U-Pu-Zr with minor actinide additions to verify that performance is within established database for U-Pu-Zr	2	M	40
					F.M.2.1b	Transient irradiation testing of U-Pu-Zr with minor actinide additions under selected DBA conditions to verify that performance is within established database for U-Pu-Zr	2	M	Incl. In Safety
	F.M.2.2	Determination of burnup and operating limits for U-Pu-Zr	P	3	F.M.2.2	Steady-state irradiation testing of U-Pu-Zr at 2-sigma power and temperature conditions	2	M	40
	F.M.2.3	Technology to mitigate fuel/cladding interdiffusion	P	2	F.M.2.3	Development of cladding liners or other concepts to fuel-cladding constituent interdiffusion	2	M	20
	F.M.2.4	Cladding materials with high-temperature properties that are improved over those of HT9-like ferritic martensitic stainless steels	P	3	F.M.2.4a	Identification and out-of-pile testing of candidate cladding and duct alloys with improved high-temperature strength and creep resistance	3	M	30
	F.M.3	Modeling and Code Development							
	F.M.3.1	Models to demonstrate understanding of fuel/cladding interdiffusion and fuel constituent migration	V	3	F.M.3.1a	Development of model for fuel/cladding interdiffusion	3	S	2
					F.M.3.1b	Development of model for fuel constituent migration	3	S	1
	F.M.3.2	Development of fuel performance models	P	2	F.M.3.2	Development of models of other (non-FCI, non-FCM) fuel performance phenomena	2	M	2

	Technical gap/issue				R&D Items				
Sub-System	Gap Label	Brief Description of Gap/Issue	Signific. of Gap *(a)*	Current TRL *(b)*	Activity Label	Brief Description of R&D Activity	Priority *(c)*	Time *(d)*	Estimated Cost Range (Million USD) *(e)*
	F.M.3.3	Modernization of LIFE-METAL	P	1	F.M.3.3	Updating of LIFE-METAL computational schemes and incorporation of new fuel performance models	2	M	5
Fuel (Oxide)	F.O.1	Property Determination							
	F.O.1.1	Thorough evaluation of temperature-dependent thermpphysical properties	P	3	F.O.1.1	Measurement of temperature-dependent thermophysical properties of(U,Pu)O2 (with minor actinide additions)	2	M	2
	F.O.1.2	Thorough assessment of phase equilbria in the (U,Pu)O2 system with minor actinide additions	P	3	F.O.1.2	Phase identification and stability determination for important temperature and composition ranges	3	M	2
	F.O.2	Irradiation Performance							
	F.O.2.1	Irradiation performance of fuel with minor actinide additions	P	3	F.O.2.1a	Steady state irradiation testing of (U,Pu)O2 with minor actinide additions to verify that performance is within established database for MOX	2	M	50
					F.O.2.1b	Transient irradiation testing of (U,Pu)O2 with minor actinide additions under selected DBA conditions to verify that performance is within established database for MOX	3	S	Incl. In Safety
	F.O.2.2	Cladding materials with high-temperature properties that are improved over those of HT9-like ferritic martensitic stainless steels	P	3	F.O.2.2a	with improved high-temperature strength and creep resistance	3	S	20
					F.O23.2a	Development of welding/joining techniques for application to selected ODS alloys	3	S	3
					F.O.2.2a	Irradiation testing of selected ODS alloys	3	S	20
	F.O.3	Modeling and Code Development							
	F.O.3.1	Establish fuel and safety performance models for application to minor actinide-bearing MOX fuel	P	4	F.O.3.1	Evaluate fuel and safety performance models for application to minor actinide-bearing MOX fuel, and develop new models as necessary	3	M	5
	F.O.3.2	Development of cladding performance models	O	4	F.O.3.2	Development of performance models for advanced (ODS) cladding materials and their incorporation into fuel performance codes	2	ML	2

See the first table in this Appendix for explanation of the footnote legend
Footnotes (e) and (f) do **not** apply

TWG 3 (L1/L2)
R&D Scope for Section 3.4 (Reactor Technology)
Version 073102

	Technical gap/issue				R&D Items				
Sub-System	Gap Label	Brief Description of Gap Issue	Signific of Gap *(a)*	Current TRL *(b)*	Activity Label	Brief Description of R&D Activity	Priority (c)	Time (d)	Estimated Cost Range (Million USD)
Reactor Technology	R4.1	Improvement of ISI&R technologies to confirm the integrity of under-sodium safety related structures and boundaries, and to repair structures in-place quickly	V	3,4	R1a	Development of high quality sensors under 200deg.C sodium	2	L	
					R1b	Development of accurate remote handling systems such as manipulators that are mobile in narrow space	2	L	
					R1c	Development of high resolution and quick image processing systems	2	L	20-50
	R4.2	Development of steam generators with high reliability	V	3,4	R2a	Development of the earlier detection system against small leak	2	L	
					R2b	Acquisition of wastage data and high temperature creep strength data for high Chromium steel	2	M	
					R2c	Development of the comprehensive evaluation method for tube rupture propagation behavior	2	L	
					R2d	Demonstration of self-limiting behavior of tube ruptures by experiments	3	L	
					R2e	Investigation of the double-wall-tube SGs with secondary system	3	L	
					R2f	Development of new steam generator concepts, where the possibility of sodium water reaction would be ruled out	3	VL	50~100
	R4.3	Development and/or selection of structural material, components and piping (1)	V	3,4	R3a	Accumulation of material strength database for base metal	1	M-L M-L	
					R3b	Accumulation of material strength database for welded joints	1	S-M S	
					R3c	Improvement of toughness and ductility	1		
		Development and/or selection of structural material, components and piping (2)	P	3,4	R3d	Establishment of welding procedure optimized for nuclear class materials	1	L	
					R3e	Development and verification of vibration response analysis model for integrated components (IHX + primary pump)	1	M	
		Development and/or selection of structural material, components and piping (3)	P	3,4	R3f	Acquisition of specific wear rate of 12Cr-steel for integrated components (IHX + primary pump)	1	M-L	
					R3g	Selection of the best 3-D seismic isolation system between 3-D isolated building system or vertically isolated main components with horizontally isolated building system	3	M-L	
					R3h	R&D for the 3-D seismic isolation device and demonstration of the performance.	3		50-100

See the first table in this Appendix for explanation of the footnote legend
Footnotes (e) and (f) do **not** apply

TWG 3 (L6/L4)
R&D Scope for Section 4 (Lead-cooled systems)
Version 073102

	Technical gap/issue				R&D Items				
Sub-System	Gap Label	Brief Description of Gap/Issue	Signific. of Gap *(a)*	Current TRL *(b)*	Activity Label	Brief Description of R&D Activity	Priority *(c)*	Time *(d)*	Estimated Cost Range (Million USD) *(e)*
Fuel/Clad/ Coolant Performance Key Viability Pb-Bi at 550^0C	A1	Pb-Bi Coolant/Ferritic Martensitic Clad Compatibility at 550C and Coolant Chemistry Control	V	2	A1a	Oxygen Sensors/Oxygen Control Technology Development including irradiated environment life tests; static and dynamic	1	M	2.5
					A1b	Endurance testing of candidate clads including alloy options, cooling options, temperature range (dynamic and static) and flow velocity	2	L	2.0
					A1c	Extension of Oxygen Sensor and Coolant Control Technologies to Pool Layouts and/or Natural Circulation	2	L	4.0
	A2	U/TRU/Zr Alloy Fuel/Pb-Bi Compatibility and/Clad Compatibility	P	1	A2a	Metal fuel/Pb-Bi Solubility/Mass Transport Tests (Run Beyond Cladding Breach)	1	S	1
					A2b	Unirradiated capsule heat soak tests of as fabricated pins in Pb-Bi at 550^0C, 570^0C, 600^0C, 625^0C	2	M	3
					A2c	Irradiated capsule tests of as fabricated pins in Pb-Bi at 550^0C	2	L	5
	A3	TRU/Nitride Fuel/Pb-Bi Compatibility and Clad/Compatibility	V	1	A3a	U Nitride Thermal/Chemical/Physical Properties Database	1	M	1.5
					A3b	Fabrication Technology Screening (iteratively with A3a and A3c)	1	M	10
					A3c	TRU Nitride Thermo/Chemical/Physical Properties Database	1	M	1.5
					A3d	(Same as A2a)	1	S	1
					A3e	(Same as A2b)	2	M	3
					A3f	(Same as A2c)	2	L	5
	A4	Th/U/TRU/Zr Alloy Fuel/Pb-Bi Compatibility and Clad/Compatibility	(TBD)		A4a-f	(Same as A3a-f)			
Fuel/Clad/ Coolant Performance Long Lead Pb at 800^0C	A5	Pb Coolant @ 800^0C Clad Material Compatibility and Coolant Chemistry Control Regime Precursors: None	V	1	A5a	Corrosion/Mass Transport Screening of Candidate Materials	1	M	3
					A5b	Clad Fabrication Screening of Materials	1	M	2
					A5c	Clad Physical properties Database Development Screening Phase Post downselection Phase	1	L	1.5
					A5d	Coolant Chemistry Control R&D	1	M	0.5

	Technical gap/issue				R&D Items				
Sub-System	Gap Label	Brief Description of Gap/Issue	Signific. of Gap *(a)*	Current TRL *(b)*	Activity Label	Brief Description of R&D Activity	Priority *(c)*	Time *(d)*	Estimated Cost Range (Million USD) *(e)*
	A6	TRU Nitride Fuel/Clad/Pb Compatibility at 800^0C Pb Precursors: A3, A5			A6a	As fabricated TRU Nitride Physical Properties Database Development (done in A3a + A3c)	1	M	0
					A6b	Unirradiated capsule heat soak tests of as-fabricated pins in Pb at 800^0C	1	M	1
					A6c	Irradiation capsule tests of as fabricated pins in Pb at 800^0C	1	L	3
Neutronics and Control Key Viability	B1	Basic Nuclear Data	V	3	B1a	Re-evaluations of Pb-Bi Minor Actinide Basic Nuclear Data	1	S	0.2
					B1b	Conduct computational benchmarks tied to existing integral experiment database - including validation of neutronics design codes for open (leaky) lattice and heterogeneous lattice	1	M	0.5
	B2	Thermostructural Reactivity Feedbacks	V	2	B2a	Develop a passive load follow/passive safety control strategy by design analysis	2	S	0.5
					B2b	Thermo/structural design optimization of ductless, open lattice core clamping approach to achieve favorable power coeff of reactivity (to enable passive load follow/passive safety)	2	M	1.0

See the first table in this Appendix for explanation of the footnote legend
Footnotes (e) and (f) do **not** apply

TWG 3 (L6/L4)
R&D Scope for Section 4 (Lead-cooled systems)
Version 073102

	Technical gap/issue				R&D Items				
Sub-System	Gap Label	Brief Description of Gap/Issue	Signific. of Gap *(a)*	Current TRL *(b)*	Activity Label	Brief Description of R&D Activity	Priority *(c)*	Time *(d)*	Estimated Cost Range (Million USD) *(e)*
Lattice Heat Removal in Pb-Bi Lattices at 500^{0}C	C1	Heat Transfer and Pressure Drop Correlations	V	2	C1a	Pb-Bi at 500^{0}C Open Lattice: Grid Spacers	2	M	5
					C1b	Forced Flow Δp { correlations heat transfers { correlations - Natural Circulation Flow Δp { correlations heat transfers { correlations	1	M	5
Key Viability					C1c	Redistribution flows in ductless assemblies	1	M	1
					C1d	Repeat C1a, C1b, C1c for IHX geometries	1	M	8
	C2	Develop and Validate Thermal/Hydraulics Modeling Codes suited for the open lattice/ductless assembly HLMC situation	P	2	C2a	Develop Transient Computational Models of Open Lattice/Grid Spacer/Ductless assemblies which couple fluid heat transfer and pressure drop to fuel pin heat transfer- representing the core itself	2	L	5
					C2b	Validate it against a broad range of test results	2	L	1.5
Heat Transport and					C2c	Couple the core code to an overall primary heat transport circuit transient thermal/hydraulics modeling	2	L	incl. above
					C2d	Validate	2	L	incl. above

	Technical gap/issue				R&D Items				
Sub-System	Gap Label	Brief Description of Gap/Issue	Signific. of Gap *(a)*	Current TRL *(b)*	Activity Label	Brief Description of R&D Activity	Priority *(c)*	Time *(d)*	Estimated Cost Range (Million USD) *(e)*
Components	D1	Heat Exchanger Materials for 800^0C Service	V	1	D1a	Identify Candidate Materials	1	M-L	15
						- In Pb/He			
						In Pb/CO_2			
Viability						In Pb/Molten Salt			
						- In He/HBr + Steam			
						In CO_2/Hbr + Steam			
						In Salt/HBr + Steam			
					D1b	Conduct Corrosion/Mass Transport Screening	incl. above	incl. above	incl. above
						- Materials Properties Screening			
						- Fabrication Screening			
					D1c	Do properties testing of downselected materials	1	L	2
					D1d	Code case development	2	VL	0.5
	D2	Pool Flows	V	2	D2a	Get Δp form factors for change in area entrance and exit losses (forced and natural circulation)	2	M-L	10
					D2b	Scale Testing of Stagnant Zones/Thermal stratification vs transient flow condition and initial condition	incl. above	incl. above	incl. above
	D3	Lift Pump Development	P	1	D3a	Lift pump bubbly flow regimes/slip ratios	3	M-L	8
					D3b	Gas/liquid separators	incl. above	incl. above	incl. above
					D3c	Blower Designs	3	M-L	5
						Temperature Increase			
						Tolerance to Pb carryover			
	D4	Direct Contact Heat Exchange	O	2	D4a	Bubbly flow regimes/slip ratios heat transfer coefficients	3	M-L	10
					D4b	Stable Evaporation/Superheat/Pressure Correlations	incl. above	incl. above	incl. above
					D4c	Injector Designs	incl. above	incl. above	incl. above
					D4d	Gas/Liquid Separators; avoidance of vapor dragover to	1	M	5
						downcomer			
	D5	Close Coupled IHX or SG	O	1	D5a	Survey the published concepts and ongoing R&D	2	S	0.5
					D5b	Small scale testing of concepts	3	L	15

TWG 3 (L6/L4)
R&D Scope for Section 4 (Lead-cooled systems)
Version 073102

	Technical gap/issue				R&D Items				
Sub-System	Gap Label	Brief Description of Gap/Issue	Signific. of Gap *(a)*	Current TRL *(b)*	Activity Label	Brief Description of R&D Activity	Priority *(c)*	Time *(d)*	Estimated Cost Range (Million USD) *(e)*
Structures, Shielding, and Refueling	E1	Seismic Isolation Development	V	2	E1a	2D, 3D Design Strategies for Seismic Isolators	2	M-L	10
					E1b	Displacement-Compliant Piping Strategies	2	M-L	10
	E2	High Temperature Concrete	P	2	E2a	Develop High Temperature Concrete Structural Designs for Silos, for Reactor Vessels	3	M-L	10
Viability					E2b	Develop and Approve High Temperature Concrete Code Case	3	L-VL	1
	E3	Refueling			E3a	Develop Refueling/Fuel Hold Down Design Strategies for High Specific Gravity Coolants			
Long Lead	E5	High temperature structural materials Pb at 800ºC Coordinate with A5 (clad-high fluence) and with D1 (HX - Corrossive fluids)	V	1	E5a	Corrosion/Mass Transport Screening	1	M	10
					E5b	Materials Properties Screening	incl. above	incl. above	incl. above
					E5c	Fabrication Screening	incl. above	incl. above	incl. above
					E5d	Coatings Screening	incl. above	incl. above	incl. above
					E5e	Downselections and initiate endurance testing	1	M-L	5
					E5f	Develop and approve High Temperature Code Case	1	L	0.5
	E6	Fabrication Development of High Temperature Structures Precursors: E5 (especially E5c)	V	2	E6a	Survey aerospace composite fabrication technology	1	S	1.5
					E6b	Survey High Temperature Chemical Commodity Industry Plant Construction/Fabrication Technologies	incl. above	incl. above	incl. above
					E6c	Survey Metallurgical Refining and Glass Industry Plant Construction/Fabrication Technologies	incl. above	incl. above	incl. above
					E6d	Develop Fabrication/Construction Strategies for Vessel, Internals, Cover Deck, Refueling Apparatus	2	M-L	8-10
					E6e	Pre-Prototype Testing of Fabrication/Construction strategies	2	L	5

See the first table in this Appendix for explanation of the footnote legend
Footnotes (e) and (f) do **<u>not</u>** apply

TWG 3 (L6/L4)
R&D Scope for Section 4 (Lead-cooled systems)
Version 073102

	Technical gap/issue				R&D Items				
Sub-System	Gap Label	Brief Description of Gap/Issue	Signific. of Gap *(a)*	Current TRL *(b)*	Activity Label	Brief Description of R&D Activity	Priority *(c)*	Time *(d)*	Estimated Cost Range (Million USD) *(e)*
Overall Safety Strategy Viability		Gaps in the areas of - Passive Decay Heat Removal - Passive Self Regulation of Reactivity - Burnup Reactivity Swing - Flow redistribution Are covered already in other tasks							
	F1	Neutronics Critical Experiments	P	2	F1a	Collect Data from prior critical experiments involving Pb, Bi and minor actinides and post analyze these experiments	2	M	1.5
					F1b	Engage in International Data/Code/Modeling Benchmarks tied to the data above	incl. above	incl. above	incl. above
					F1c	Plan a Critical Experiments Program and Prepare Facility, Materials, and Project Team	3	L-VL	25
	F2	Fuel/Clad/Coolant Phenomenology Under Severe Accident Conditions	V	1-2	F2a	Add Initiating Phase and Transition Phase Models to Accident Analysis Codes for Postulated Phenomenology based on single effects tests and literature data	2	L	5
					F2b	Evaluate consequences by analysis	incl. above	incl. above	incl. above
					F2c	Plan Test Program of inpile transient tests on single pins and pin clusters in flowing Pb	3	VL	20
	F3	Invessel Steam Generator or IHX Tube Rupture	V	1-2	F3a	Develop Computational Models of Rupture, Propagation, Blowdown and Entrainment	2	M	3
					F3b	Perform Mid Scale testing and calibrate computational models	2	L	5
					F3c	Evaluate Effects and Mitigation Measures on Basis of Calculational Models	2	L	1.5
					F3d	Plan Large Scale Tests; Prepare Facility Selection (or Design of New Facility)	3	VL	15
O&M Strategy Viability	G1	Measurability of Passive Safety Parameters	P	2	G1a	Develop the tech spec procedure to assure reactivity feedbacks are in the required range and demonstrate efficacy using dynamic modeling	2	M	0.5
					G1b	Develop the tech spec procedure to assure decay heat removal parameters are in the required range	2	M	0.5
	G2	Remote Monitoring Technology	P	2	G2a	Monitor Developments by Others of Secure Remote Monitoring Technology	3	M	0.2
					G2b	Adapt Technology to L6 Concepts	3	L	1.0

Sub-System	Technical gap/issue				R&D Items				
	Gap Label	Brief Description of Gap/Issue	Signific. of Gap *(a)*	Current TRL *(b)*	Activity Label	Brief Description of R&D Activity	Priority *(c)*	Time *(d)*	Estimated Cost Range (Million USD) *(e)*
	G3	Polonium Management	V	2	G3a	Develop Cover Gas Management During Refueling	1	M	1.0
					G3b	Develop Po coolant and cover gas cleanup system	1	M-L	10.0
	G4	In Service Inspection	V	2	G4a	Develop remote leak detectors; crack detectors for service in Pb at >400ºC	1	L	15
					G4b	Technology Testing in Medium Scale Pb pool Facility at Temperature (tie to D2)	incl. above	incl. above	incl. above

See the first table in this Appendix for explanation of the footnote legend
Footnotes (e) and (f) do **not** apply

TWG 3 (L6/L4)
R&D Scope for Section 4 (Lead-cooled systems)
Version 073102

Sub-System	Technical gap/issue				R&D Items				
Sub-System	Gap Label	Brief Description of Gap/Issue	Signific. of Gap *(a)*	Current TRL *(b)*	Activity Label	Brief Description of R&D Activity	Priority *(c)*	Time *(d)*	Estimated Cost Range (Million USD) *(e)*
Fabricability and Capital Cost Reductions R&D	H1	Identify Vessel and Internals Materials Suitable for service conditions <u>and</u> amenable to Advanced Fabrication Techniques Coupled to D1 and E5	V	1	H1a	Survey Aerospace Materials Survey Fusion Materials Survey High Temperature Chemical Plant Materials	Covered in	D1 and E5	
					H1b	(This is coupled to Tasks A5, D1)			
					H1c	Survey Fabrication Technologies in the above industries Adapt H1a and b to the L6/L4 concept sets			
	H2	Identify Modularization/Factory Fabrication Technologies for Large Civil Construction	V	2	H2a	Survey Technologies in Shipyards, Ocean Oil Rigs, Long-Span. Bridges, etc. (including Japanese and Korean Nuclear Power Plants)	1	S-M	1
	H3	Identify Integrated Design/Fabrication/Transport/ Installation/Startup Process Planning Supporting Software and Document Change Control Processes	P	2	H3a	Survey Technologies in Industries named in H2a	2	M	0.5
	H4	Lay out a strategy with integrates H1, H2, H3 and do cost analyses	P	2	H4a	Define a sample concept for exercising the integrated strategy Apply the approaches learned/developed above Estimate overnight cost	2	M-L	1.0
Fuel Cycle R&D Nitride Pyro	I1	Nitride Pyro Recycle	V	1	I1a	Monitor Metal and Oxide Pyro Development	1	M-L	25
					I1b	Single Effects tests and Screening of Options		↓ ↓	
					I1c	Nitrogen Recovery Technique Development		then	
					I1d	In Situ Renitriding Development		↓ L-VL	
					I1e	Flowsheet Design			
					I1f	Bench Scale Testing	1		30
	I2	N15 Enrichment	V	1	I2a	Evaluation and Screening of Options	1	S	1
					12b	Small Scale Testing	1	M	5
					12c	Economic Evaluations of Options	1	M-L	3
	I3	Pyro Waste Form Production	P	2	13a	Monitor Metal and Oxide Pyro Waste Form	2	L-VL	10
					13b	Developments Adapt to nitride	incl. above	incl. above	incl. above

	Technical gap/issue				R&D Items				
Sub-System	Gap Label	Brief Description of Gap/Issue	Signific. of Gap *(a)*	Current TRL *(b)*	Activity Label	Brief Description of R&D Activity	Priority *(c)*	Time *(d)*	Estimated Cost Range (Million USD) *(e)*
	I4	Post Pyro Remote Refabrication	V	1	14a	Crushing Development	1	M-L	15
		Couple to A3, A6			14b	Blending/Composition Control		↓	
					14c	Bonding (if required)		then	
					14d	Packing/Compaction		↓	
					14e	Quality Confirmation and Inspection	3	L-VL	25
	I5	Safeguards	P	2	15a	Develop Integrated Safeguards Regime	2	L-VL	10
					15b	Develop Remote Sampling of Radioactive Material	incl. above	incl. above	incl. above
					15c	Develop Remote Accountancy of Radioactive Material			

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TWG 3
R&D Scope for Section 4 (Lead-cooled systems)
Version 073102

	Technical gap/issue				R&D Items				
Sub-System	Gap Label	Brief Description of Gap/Issue	Signific. of Gap *(a)*	Current TRL *(b)*	Activity Label	Brief Description of R&D Activity	Priority *(c)*	Time *(d)*	Estimated Cost Range (Million USD) *(e)*
Nitride Aqueous	I6	Aqueous Nitride Recycle	(TBD)		16a	Monitor Advanced Aqueous for Oxide			
					16b	Develop Front end Processes including Nitrogen recovery			
	I7	Aqueous Waste Form Production	(TBD)		17a	Monitor Advanced Aqueous Oxide Waste Form Work			
					17b	Adaptations to Nitride			
	I8	Post Aqueous Remote Refabrication	(TBD)		18a	Monitor Gelation Work for Oxide			
					18b	Monitor N15 Enrichment Work			
					18c	Screen and downselect Gelation and Renitridation Process and Conduct Bench Scale Tests (Glove box w/minor actinide containing fuel)			
					18d	Bonding (if required)			
					18e	Packing/Compaction Bench Scale Testing			
					18f	Quality Confirmation/Inspection			
	I9	Safeguards	(TBD)		19a	Develop Integrated Safeguards Regime			
					19b	Develop Remote Sampling of Radioactive Material			
					19c	Develop Remote Accountancy of Radioactive Materials			
Energy Converters	J1	Supercritical CO2 Brayton Cycles	V	1	J1a	Thermodynamic Cycle Optimization	1	S	0.5
					J1b	Materials Selections - Heat Exchangers - Recuperator - Turbine and Turbine Blacks	1	M	5
					J1c	Small Scale Testing - Turbine - Recuperator	1	M-L	2
	J2	Ca-Br Water Cracking	V	1	J2a	Materials Selections (see Task D1)	1	S-M	8
					J2b	Ca Support Selection	incl. above	incl. above	incl. above
					J2c	Thermo/Chemical Properties Measurements and Database	incl. above	incl. above	incl. above
					J2d	Rate Constant Measurements	incl. above	incl. above	incl. above
					J2e	Thermodynamic Optimization and Flowsheet	incl. above	incl. above	incl. above
					J2f	Bench Scale Integral Test	incl. above	incl. above	incl. above
					J2g	Small Scale Prototype Test	1	M-L	10
	J3		P	2	J3a	Review Fossil Plant Experience Base	3	L-VL	5
					J3b	Monitor Work by Others on Sc Steam Rankine Cycle TWG-1 Russian BREST Program	incl. above	incl. above	incl. above
					J3c	Economic Comparisons of J1 vs J3	incl. above	incl. above	incl. above

Sub-System	Technical gap/issue				R&D Items				
	Gap Label	Brief Description of Gap/Issue	Signific. of Gap *(a)*	Current TRL *(b)*	Activity Label	Brief Description of R&D Activity	Priority *(c)*	Time *(d)*	Estimated Cost Range (Million USD) *(e)*
	J4	Desalinization	P	4	J4a	Develop Models/Adapt IAEA Model for Nuclear Desalinization	3	M-L	2
					J4b	Monitor R&D Progress by Others on Reverse Osmosis and Multi-effects Distillation	incl. above	incl. above	incl. above
					J4c	Monitor developments by Other of Multi-Stage Flash Heat Exchangers, Crud Control and Brine Disposition	incl. above	incl. above	incl. above
					J4d	Evaluate Commercial Opportunities for Coupling to Product Extraction from Brine - Uranium - Other	3	M-L	1

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